PSNH PUBLIC SERVICE
Company of New Hampshire

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November 4, 1983

SBN- 576
T.F. B4.2.99

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated July 8, 1983, "Required Action Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28)", D. G. Eisenhut to All Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits

Subject: Response to Generic Letter 83-28

Dear Sir:

In response to Generic Letter 83-28, in which you requested information pursuant to 10CFR50.54(f), we have enclosed a report which provides the status of our current conformance with the positions contained therein and our plans and schedules for any needed improvements for conformance with the positions.

As indicated in the response to numerous items, additional information will be forthcoming.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John DeVincentis

John DeVincentis
Project Manager

ALL/dsm
Enclosure
cc: Atomic Safety and Licensing Board Service List

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1.1 POST TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)	
POSITION	RESPONSE
1.1. Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely. A report describing the program for review and analysis of such unscheduled reactor shutdowns should include, as a minimum: 1.1.1. The criteria for determining the acceptability of restart. 1.1.2 The responsibilities and authorities of personnel who will perform the review and analysis of these events. 1.1.3 The necessary qualifications and training for the responsible personnel. 1.1.4 The sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to re-construct the event accurately and in sufficient detail for proper understanding. (See Action 1.2) 1.1.5 The methods and criteria for comparing the event information with known or expected plant behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function). 1.1.6 The criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Plant Operations Review Committee, will be consulted prior to authorizing restart) and	1.1 A Post Trip Safety Assessment procedure is planned for completion 3 months prior to fuel load of Seabrook Station Unit I. The procedure, as planned, will address the requirements of NRC generic letter 83-28, Post Trip Review Position and will consider along with other pertinent information, the guidelines provided by INPO Good Practice OP-211 (Draft dated 9/83 or its' successor) entitled Post Trip Review. In concert with procedural controls, final software for computerized data presentation is also being developed.

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1.1 POST TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)	
POSITION	RESPONSE
(Cont'd) guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event. 1.1.7. Items 1 through 6 above are considered to be the basis for the establishment of a systematic method to assess unscheduled reactor shutdowns. The systematic safety assessment procedures compiled from the above items, which are to be used in conducting the evaluation, should be in the report.	

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1.2 POST TRIP REVIEW - DATA AND INFORMATION CAPABILITY	
POSITION	RESPONSE
1.2 Licensees and applicants shall have or have planned a 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. Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Action 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures. A report shall be prepared which describes and justifies the adequacy of equipment for diagnosing an unscheduled reactor shutdown. The report shall describe as a minimum:	1.2 The major source of data for the systematic diagnosis of the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment is the Main Plant Computer System (MPCS).
1.2.1 Capability for assessing sequence of events (on-off indications). 1. Brief description of equipment (e.g., plant computer, dedicated computer, strip chart) 2. Parameters monitored 3. Time discrimination between events 4. Format for displaying data and information	1.2.1 The MPCS consists of an integrated network of twenty-two computers. Two large host computers are located within the control room complex. Both host computers are connected via data links to ten Intelligent Remote Terminal Units (IRTU's) which are located in areas of the plant convenient to instrumentation concentrations. Each IRTU contains two smaller computers and the process input-output hardware necessary to sample plant system related parameters and status changes.

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1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY	
POSITION	RESPONSE
(Cont'd) 5. Capability for retention of data and information 6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable)	(Cont'd) The MPCS configuration (Figure 1) was selected with reliability as the primary goal. Each computer and peripheral device has a redundant backup device capable of assuming all real time processing functions. In addition, any failed component, with the exception of the process I/O hardware, can be isolated, repaired, and returned to service without degrading system operations. MPCS power is supplied to each redundant Host Computer by two individual non-class IE uninterruptible power supplies (UPS). Normal power for each UPS is supplied at 480 VAC from an emergency diesel generator protected vital AC bus. Normal power is first rectified, then inverted to produce a regulated 120/208 VAC output to supply the MPCS (Figure 2). A separate battery backup source is supplied to each UPS for a period of fifteen minutes following a normal power loss. Additional protection is provided for UPS failure by a separate unregulated 480 VAC/230 VAC vital bus supply which is automatically connected by an internal solid state automatic transfer switch to assure a continuous source of power for the MPCS. The sequence of events data is recorded in a disc based file capable of storing 50,000 change of state events. Events recorded in the Digital Archive System consist of three general types of state changes. The first presents information generated by field contact status changes, the second presents selected status changes generated when computer sampled process variables exceed preset limits, and the third is generated when selected calculated variables exceed preset limits.

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1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY	
POSITION	RESPONSE
	(Cont'd) The Digital Archive System records all state changes indicated above. The selection of 50,000 event file size assures that a significant event history exists prior to an unusual event to which a systematic safety assessment may apply. Digital Archive System report generation is initiated from the main control board through the use of a CRT based man-machine interface. The requestor inputs the starting time and date, and the ending time and date to initiate report generation. The report generated is sequential in time and is accurate to within 10 milli-seconds. Changes of state for both set and reset events are recorded in individual columns. A sample report is provided as attachment 1. Approximately two thousand-five hundred identified digital inputs are provided. a list of these data points, and those noted on the next page, is available at the Station.
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. 1. Brief description of equipment (e.g., plant computer, dedicated computer, strip charts) 2. Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate	1.2.2 Approximately one-thousand-three hundred analog values and about one-thousand-three hundred analog engineering values are also processed by the MPCS. The analog data values are archived in three circular disc files formats. A five second file format saves each five second data value in a circular disc file sized for a one hour retention. A five minute disc file saves the engineering values in five minute increments for a period of seven days. Finally, an hourly sample of the five minute file is maintained on a thirty-six day circular disc file which is subsequently saved on tape as a permanent plant record.

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY	
POSITION	RESPONSE
(Cont'd)	(Cont'd)
3. Duration of time history (minutes before trip and minutes after trip)	
4. Format for displaying data including scale (readability) of time histories	
5. Capability for retention of data, information, and physical evidence (both hardware and software)	
6. Power source(s) (e.g., Class IE, non-Class IE, non-interruptable)	
1.2.3 Other data and information provided to assess the cause of unscheduled reactor shutdowns.	1.2.3 In addition to the MPCS supplied data, the following systems and components provide additional information which may be used in support of post-event analysis. At present there are, in addition to the computer records, 54 Class IE Post Accident Monitoring recorders. These recorders provide time histories of appropriate plant parameters in a strip chart format.
1.2.4 Schedule for any planned changes to existing data and information capability.	1.2.4 Analog archive data retrieval software is planned to produce hardcopy outputs which are formatted to support systematic safety assessment by appropriately qualified individuals. Radiation historical information, storage and hardcopy will be available from the Radiation Data Monitoring System (RDMS). The RDMS is an arrangement similar to the MPCS in that two redundant real time host computers address field mounted micro-processors. The RDMS normal data storage will be provided

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1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY	
POSITION	RESPONSE
	(Cont'd) on a hard disc storage medium for eventual transfer to magnetic tape for long term record retention. The field mounted microprocessors are provided with battery backup to provide up to eight hours of continued service. Each individual microprocessor is to have a data storage period of twenty-four hours independent of the host computer units. Software is planned which will provide hardcopy printouts in a format compatible with systematic analysis.

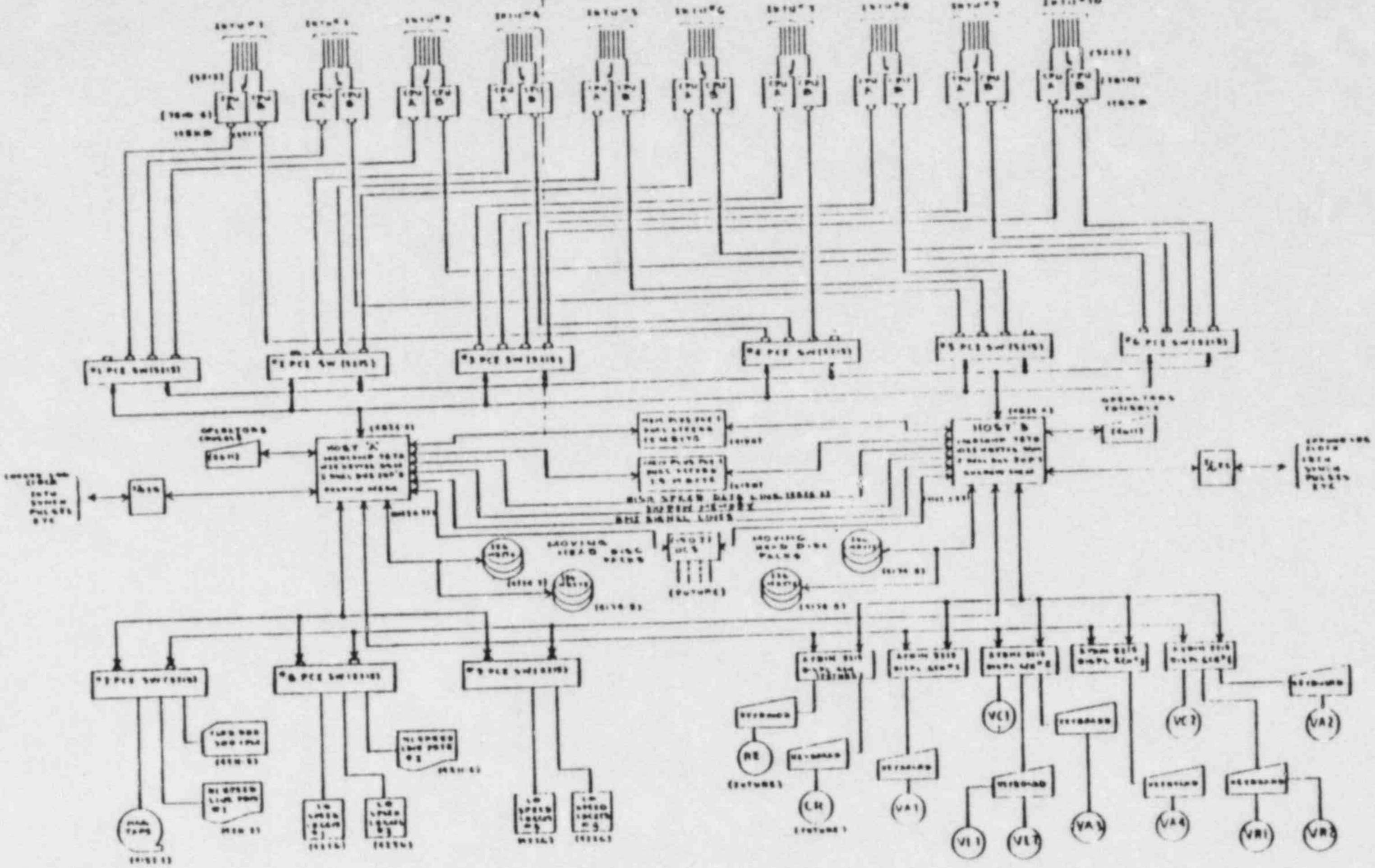

FIGURE 1. MPCS HARDWARE CONFIGURATION

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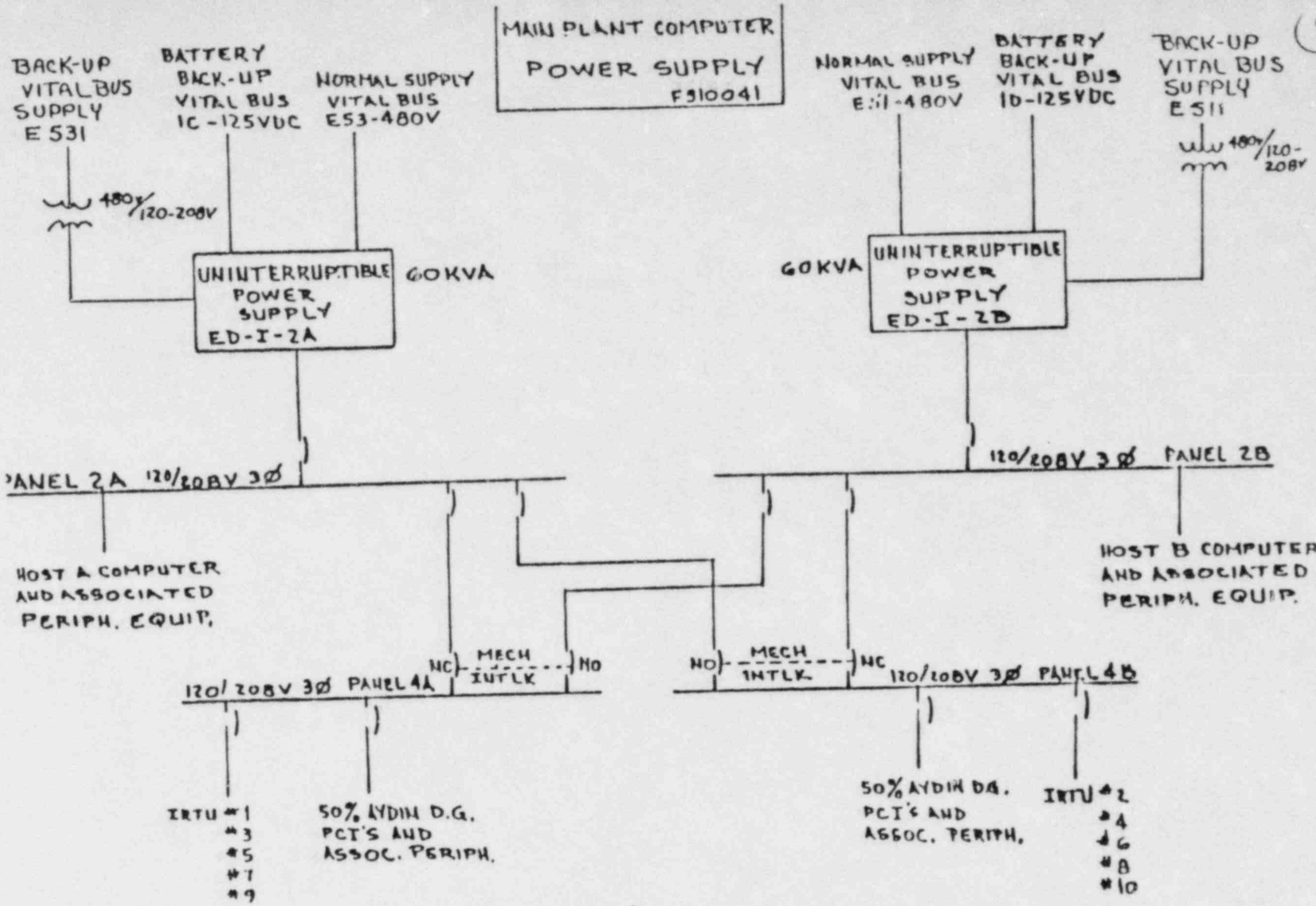

ELECTRICAL DISTRIBUTION SYSTEM
FIGURE 2

DATE 9/23/85 VIDEO ALARM ARCHIVE REVIEW MPCS-UNIT 1 PAGE 1

11:21:45:910	6394	BUS E-61 FOR BKR CTL	REMOTE
11:24: 9:890	6394	BUS E-61 FOR BKR CTL	REMOTE
11:24:53:570	6394	BUS E-61 FOR BKR CTL	REMOTE
11:27:17:270	63BB	480 V BUS E53 FOR BKR CTL	REMOTE
11:27:25:890	63BB	480 V BUS E53 FOR BKR CTL	REMOTE
11:27:28:580	63BB	480 V BUS E53 FOR BKR CTL	REMOTE
11:27:29:150	63BB	480 V BUS E53 FOR BKR CTL	REMOTE
11:27:35:590	63BB	480 V BUS E53 FOR BKR CTL	REMOTE
11:28:25:700	63BB	480 V BUS E53 FOR BKR CTL	REMOTE

11:21:44:720	6394	BUS E-61 FOR BKR CTL	LOCAL
11:24: 9:710	6394	BUS E-61 FOR BKR CTL	LOCAL
11:24: [illegible]:940	6394	BUS E-61 FOR BKR CTL	LOCAL
11:27:17:260	63BB	480 V BUS E53 FOR BKR CTL	LOCAL
11:27:17:350	63BB	480 V BUS E53 FOR BKR CTL	LOCAL
11:27:20:560	63BB	480 V BUS E53 FOR BKR CTL	LOCAL
11:27:28:970	63BB	480 V BUS E53 FOR BKR CTL	LOCAL
11:27:[illegible]	63BB	480 V BUS E53 FOR BKR CTL	LOCAL
11:27:35:400	63BB	480 V BUS E53 FOR BKR CTL	LOCAL

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Attachment 1

311:690

2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)	
POSITION	RESPONSE
2.1.1 Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement.	2.1.1. Reactor Trip System components will be identified as safety-related in those station administrative systems that control safety-related activities. This effort will be completed 3 months prior to fuel load. Procurement of spare parts, for Reactor Trip System Components is coordinated to insure the qualification of equipment is maintained or increased. Existing Seabrook Station Administrative and QA programs provide the necessary instructions and guidelines required to identify Nuclear Safety Class components and subcomponents into the spare parts program. Corrective and preventative maintenance activities are performed in accordance with approved work control program procedures. These procedures require the use of work documents which receive appropriate review and appro- prior to commencement of work and clearly identify the nuclear safety nature of the work.
2.1.2 In addition, for these components, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures.	2.1.2. This requirement is satisfied by existing provisions of the Seabrook Station Administrative and QA Program in that: The Seabrook Station administrative program/procedure requires that maintenance or operating procedures cite in their references the appropriate vendor information, manuals and other instructions.

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2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)	
POSITION	RESPONSE
	Within a procedure itself the vendor information is used, as needed and in combination with other pertinent technical data, test data, utility knowledge and industry experience, in the drafting, engineering/technical review and quality review and approval of the safety-related procedure. Existing Seabrook Station administrative procedures provide control of incoming vendor data, whether it arrives direct from the vendor or from other industry or regulatory sources (i.e., NOTEPAD, NPRDS, SEE-IN, NRC Bulletins, etc), such that it receives the appropriate engineering/technical screening, review and distribution for immediate notification or other timely incorporation into station maintenance or operating procedures, equipment data records, training programs or future review and revision cycles. The incorporation of such safety-related information (or changes) remains within the scope and requirements of the QA program review and approval system.
2.1.3. Vendors of these components should be contacted and an interface established. Where vendors can not be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability.	2.1.3 This requirement is satisfied by existing provisions of the Seabrook Station Administrative and QA Program in that: The Station maintenance or operating procedures that cannot cite valid vendor information were drafted with due regard for this lack of data and incorporate appropriate technical and engineering information and directions based on review and consideration of other available information, such as NPRDS, SEE-IN, NOTEPAD, NOMIS, NRC Bulletins, test data, utility knowledge and industry experience.

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2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)	
POSITION	RESPONSE
	Existing Seabrook Station administrative procedures provide control of incoming technical and engineering data, whether it is from a vendor or from other industry or regulatory sources (i.e., NOTEPAD, NPRDS, SEE-IN, NRC Bulletins, etc.), such that it receives the appropriate engineering/technical screening, review and distribution for immediate notification or other timely incorporation into station maintenance or operating procedures, equipment data records, training programs or future review and revision cycles. The incorporation of such safety-related information (or changes) remains within the scope and requirements of the QA program review and approval system.
2.1.4. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings.	2.1.4. This requirement is satisfied by existing provisions of the Seabrook Station Administrative and QA Program in that: Existing station procedures provide necessary control of incoming vendor information. The station is presently and will continue to maintain periodic communication with the Reactor Trip System (NSSS) supplier as defined in Section 2.1.3.
2.1.5 The program shall also define the interface and division of responsibilities among the licensees and the nuclear and nonnuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.	2.1.5 This requirement is satisfied by existing provisions of the Seabrook Station Administrative and QA Program in that: The services of the vendor/contractor who will perform safety-related services must first be an approved/qualified supplier of nuclear safety-related services. Furthermore, these services are specified in the Purchase Order documentation that establishes, depending on the circumstances, one or more of the following combination of controls:

2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)	
POSITION	RESPONSE
	1. The service is performed using the station safety-related procedures that have been approved after a technical and quality review cycle typical for other station service, maintenance, repair or operating procedures. - OR - 2. The service is performed using the vendor/contractor safety-related procedures that have been reviewed and approved in accordance with Seabrook Station procurement program, QA program and administrative review program, such that these vendor/contractor documents are processed and approved in a manner equivalent to station safety-related procedures concerning similar activities. - AND - 3. The activity will be performed under the cognizance of the Seabrook Station QC/QA Program. - OR - 4. The activity will be performed under the cognizance of the vendor/contractors QC/QA program which has been separately reviewed and approved in accordance with the station QA program. In addition, during the performance of the service, the station QA program will monitor the effectiveness of the vendor/contractor performance and compliance with its approved program by suitable surveillance, inspection and audit. When the Technical Representatives from a vendor/manufacturer arrive on site to perform safety-related services, maintenance or repair activities, the spare parts they may bring with them, their tools and their troubleshooting, maintenance or repair procedure are brought into and are processed within the requirements of station Procurement Program and QA Program.

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

POSITION	RESPONSE
2.2.1.1 The criteria for identifying components as safety-related within systems currently classified as safety-related. This shall not be interpreted to require changes in safety classification at the systems level.	2.2.1.1 Components within safety-related systems are identified as safety related if it is determined that the component is required for the system to perform its safety function or if its failure could prevent the system from performing its safety function.
2.2.1.2 A description of the information handling system used to identify safety-related components (e.g., computerized equipment list) and the methods used for its development and validation.	2.2.1.2 The review of safety-related systems to identify all safety-related components is presently in progress and will be completed prior to fuel load. The program to ensure the identification of components necessary for accomplishing required safety functions is defined in section 2.1.1.
2.2.1.3 A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B, apply to safety-related components.	2.2.1.3 Existing Seabrook Station Administrative and QA Programs provide the necessary guidelines and instructions necessary for the implementation of the Maintenance and Spare Parts Programs as defined in section 2.1.1
2.2.1.4 A description of the management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed.	2.2.1.4 This requirement is satisfied by existing provisions of the Seabrook Station Administrative and QA Programs. The approved Station Manual Procedure provides the necessary management controls to insure that all station requirements have been followed. The standards prescribed in this procedure govern the preparation, review, approval, revision and maintenance of the Station Manual and its administrative procedures. Included in this procedure are selected administrative controls and quality assurance requirements which provide the station with a method for implementing directives, policies and instructions.

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

POSITICN	RESPONSE
2.2.1.5 A demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions and provide support for the licensee's receipt of testing documentation to support the limits of life recommended by the supplier.	2.2.1.5 This requirement is satisfied by the existing provisions of the Seabrook Station Administrative and QA Program. The approved station procedure for Material Purchase Request Preparation and Review provides the necessary instructions and guidelines for the purchase of safety-related components. The material purchase request will identify the nuclear safety class, required documentation, specifications, and applicable codes and regulations. A technical and quality review is performed on all safety-related material purchase requests to verify that all appropriate purchasing requirements have been specified ensuring that the Seabrook Station design and service conditions are met.
2.2.1.6 Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems, and components important to safety required by GDC-1 (defined in 10 CFR 50, Appendix A, "General Design Criteria, Introduction").	2.2.1.6 The equipment classification program for safety-related components is described in section 3.2 of the FSAR and in our response to the Quality Assurance Branch Request for Additional Information #260.28, dated August 12, 1983. In addition 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 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.
2.2.2 For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of	2.2.2 As stated in PSNH letter SBN-556 dated September 6, 1983, PSNH is participating in a NUTAC established to respond to this item. The NUTAC effort is anticipated to produce a program in May, 1984. PSNH will therefore be able to describe its program in June, 1984.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)	
POSITION	RESPONSE
2.2.2 (Cont'd) safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided	

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3.1 POST - MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)	
POSITION	RESPONSE
3.1.1 Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.	3.1.1 Draft Test and Maintenance Procedures have been written but have not yet had final Station Operational Review Committee review and approval. The SORC review is on hold pending the addition of appropriate testing requirements applicable to the final circuit design which is incorporated at Seabrook Station. The Westinghouse Owners Group has just approved a contract to design, and qualify to 1E, a trip circuit modification to wire the shunt trip into the automatic trip circuit. This contract has been awarded to Westinghouse and is scheduled for completion in 9 months. Procedures will be revised as appropriate to insure that equipment is capable of performing its safety function prior to return to service.
3.1.2 Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.	3.1.2 The vendor, Westinghouse, is designing the circuit. Station staff will review this design and include any appropriate test guidance in maintenance and post-maintenance test procedures. Testing requirements and frequencies will be added to the Technical Specifications, as appropriate, based on the incorporated design.
3.1.3 Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval. (Note that action 4.5 discusses on-line system functional testing).	3.1.3 Our review to date has found no post-maintenance testing requirements in the present draft Technical Specifications which can be demonstrated to degrade rather than enhance safety. Station Staff is in the process of reviewing and rewriting the draft Technical Specifications to conform to the format of Revision 4 of the Westinghouse Standard Technical Specification. In conjunction with this rewrite, Seabrook Station Staff is working with the Westinghouse Owners Group to identify present surveillance requirements which may be inconsistent with overall plant safety. These concerns will be identified to the NRC Task Force on Plant Technical Specifications.

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

POSITION	RESPONSE
3.2.1 Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.	3.2.1 Seabrook Stations' Work Request Procedure (AQ5.004), Repetitive Task Procedure (AQ5.005), Maintenance Program (AQ5.009) and Non-Routine Maintenance (AQ5.010) control all maintenance on safety related equipment. These general procedures already require a specific review for post-maintenance testing requirements and that such testing be included in the individual equipment procedures. Station staff is in the process of developing such procedures for specific equipment based on the requirements of the four procedures listed. These procedures are scheduled to be completed, reviewed, and implemented three months prior to fuel load.
3.2.2 Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.	3.2.2 As noted in response to 3.2.1 above, Station Staff will check vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures, or technical specifications.
3.2.3 Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.	3.2.3 Draft Test and Maintenance Procedures have been written but have not yet had final Station Operational Review Committee review and approval. The SORC review is on hold pending the addition of appropriate testing requirements applicable to the final circuit design which is incorporated at Seabrook Station. The Westinghouse Owners Group has just approved a contract to design, and qualify to 1E, a trip circuit modification to wire the shunt trip into the automatic trip circuit. This contract has been awarded to Westinghouse and is scheduled for completion in 9 months. Procedures will be revised as appropriate to insure that equipment is capable of performing its safety function prior to return to service.

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4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)	
POSITION	RESPONSE
4.1 All vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either: (1) each modification has, in fact, been implemented; or (2) written evaluation of the technical reasons for not implementing a modification exists.	4.1 The Westinghouse Owners Group has issued a project authorization for Westinghouse to develop a generic functional design for modifying the reactor trip breakers to incorporate shunt trip of the breakers via the automatic reactor protection system. Seabrook Station will modify its reactor trip breakers based on this generic design. Vendor recommended modifications as described in Westinghouse letter to the NRC, NS-EPR 2753 dated April 21, 1983 will be implemented prior to fuel loading. Vendor recommended modifications as described in Westinghouse Technical Bulletin NSD-TB-75-2 is not applicable to the DS breakers furnished for Seabrook; therefore no further action is required.

4.2 REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)	
POSITION	RESPONSE
4.2.1 A planned program or periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier.	4.2.1 A planned program of periodic maintenance has been developed. It is based on the manufacturers' Technical Manual recommendations and conforms to Seabrook Station Maintenance Program Administrataive Procedure (AQ5.009). This program includes: lubrication, housekeeping, and other items recommended by the equipment supplier.
4.2.2 Trending of parameters affecting operation and measured during testing to forecast degradation of operability.	4.2.2 A history card on each breaker will be maintained and will be periodically reviewed for trends to facilitate forecasting failures.
4.2.3 Life testing of the breakers (including the trip attachments) on an acceptable sample size.	4.2.3 The Westinghouse Owners Group has placed a contract with Westinghouse (Shop Order MUHN-2051) for cyclic life and class 1E qualification of shunt trip and cyclic testing of undervoltage trip for DS breakers.
4.2.4 Periodic replacement of breakers of components consistent with demonstrated life cycles.	4.2.4 Based on the report from Westinghouse (noted in item 4.2.3 above), periodic replacement of breaker or components will be scheduled appropriately.

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4.3 REACTOR TRIP SYSTEM RELIABILITY (AUTOMATIC ACTUATION OF SHUNT TRIP ATTACHMENT FOR WESTINGHOUSE AND B&W PLANTS)	
POSITION	RESPONSE
4.3 Westinghouse and B&W reactors shall be modified by providing automatic reactor trip system actuation of the breaker shunt trip attachments. The shunt trip attachment shall be considered safety related (Class IE).	4.3 The automatic shunt trip modification similar to that described in the generic design package submitted to the NRC by letter OG-101 on June 14, 1983 will be implemented at Seabrook. The shunt trip attachment and associated circuitry will be considered safety-related and will be qualified to meet Class IE requirements. The modification will be completed prior to fuel loading. The plant specific information requested by the Safety Evaluation Report forwarded by letter dated August 16, 1983 from D. G. Eisenhut will be submitted by July 1984.

(23)

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)	
POSITION	RESPONSE
4.5.1 The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W (see Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.	4.5.1 Station staff will incorporate testing to independently checked the shunt the undervoltage trips when the Westinghouse finalized design is available.
4.5.2 Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.	4.5.2 N/A to Seabrook Station Reactor Trip System Design.
4.5.3 Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as: 1. uncertainties in component failure rates 2. uncertainty in common mode failure rates 3. reduced redundancy during testing 4. operator errors during testing 5. component "wear-out" caused by the testing.	4.5.3 The Westinghouse Owners Group contract with Westinghouse includes requirements to prepare data during life cycle testing to determine acceptable test intervals.

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NEW ORLEANS, LOUISIANA 70174-6008 • (504) 366-2345
MIDDLE SOUTH UTILITIES SYSTEM

November 4, 1983

W3P83-3318
Q-3-A20.02.02
3-A1.01.04
I.02

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

SUBJECT: Waterford SES Unit 3
Docket No. 50-382
Generic Letter 83-28
Required Actions Based on Generic
Implications of Salem ATWS Events

REFERENCE: NRC Letter dated October 18, 1983 from G. W. Knighton
to R. S. Leddick

Dear Mr. Eisenhut:

The referenced letter requested that we transmit our response to Generic Letter 83-28 to the NRC by November 5, 1983. Accordingly, attached is the requested information.

Due to our participation in the CE Owners Group and INPO Nuclear Utility Task Action Committee, as well as the extensive nature of certain requirements, a complete status or schedule is not available for all items at this time. However, the attached response includes the status, schedule for implementation, or plans for development of a schedule for submittal of implementation dates, as appropriate, for each item in the Generic Letter. The responses are numbered in accordance with Generic Letter 83-28. Future submittals will be made on an individual item basis on the schedules provided herein.

If you have any questions, please advise.

Very truly yours,

K W Cook

K. W. Cook
Nuclear Support and Licensing Manager

KWC/MAL/pjl

cc: W. M. Stevenson, E. L. Blake, J. Wilson, G. L. Constable

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1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

LP&L is presently revising General Operating Procedure OP-10-001 to address the concerns raised by this section of Generic Letter 83-28. A report summarizing our implementation of these requirements will be submitted to the NRC by January 16, 1984.

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

1. The Sequence of Events file is scheduled to be integrated into the Plant Monitoring Computer software and tested prior to fuel load.

2. The Post Trip Review file is scheduled to be integrated into the Plant Monitoring Computer software and tested prior to fuel load.

3. The Safety Parameter Display System is scheduled to be integrated into the Plant Monitoring Computer prior to the 5% power plateau of Power Ascension Testing.

4. A report summarizing our program to implement this item will be submitted February 1, 1984.

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

LP&L is presently conducting a thorough documentation review to confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement.

As noted in Section 2.2.2 of this report, LP&L is participating in an INPO program to develop actions necessary to establish a vendor interface program. The results of the INPO program will be used to establish a near-term vendor interface program for reactor trip system components. This program may later be incorporated into the larger vendor interface program for all safety-related components on a schedule consistent with Section 2.2.2.

A final report, detailing our documentation review and vendor interface program for reactor trip system components will be submitted to the NRC by March 30, 1984.

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

2.2.1 A listing of safety-related structures, systems and components is contained in Table 3.2-1 of the Waterford FSAR. Safety-related systems are further expanded to the component level in other sections of the FSAR, for example, the master equipment list submitted in response to NUREG 0588 as referenced in FSAR Section 3.11; submittals made in accordance with seismic qualification review as referenced in FSAR Section 3.10; and the listing of safe shutdown equipment required for Appendix R and detailed in FSAR Section 9.5.1.

In response to Generic Letter 83-28 LP&L has committed to a comprehensive review of our equipment classification scheme in order to develop a refined listing of safety-related components by October, 1985.

The program that will be developed to generate this list will include: 1) criteria for identifying safety related components, 2) a system to identify safety related components using the criteria and a method for development and validation of the data, 3) procedures for use of the data by various plant groups, 4) requirements for appropriate levels of management review and approval of the program and procedure used in the list development, and 5) a priorization of systems to ensure that all safety-related components are included in the listing.

Procedures are presently in place to ensure that design verification and qualification testing are specified for procurement of safety related components. The procedures will be reviewed to ensure actions required by this section are adequately addressed.

With respect to the equipment classification program in use at Waterford 3 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 structure 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.

A report concerning our implementation of the requirements for this section will be submitted by November 15, 1985.

2.2.2 LP&L is actively participating in the INPO Nuclear Utility Task Action Committee (NUTAC) which is exploring actions that can be taken to address vendor interface problems utilizing an industry wide approach. The results of the NUTAC actions are scheduled for review and approval by member utilities in February, 1984.

Upon approval of the NUTAC results LP&L will begin development of a vendor interface program for all safety-related components. This program will be fully implemented upon completion of the safety-related equipment classification.

A description of the program and confirmation of its implementation will be provided to the NRC by November 15, 1985.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

We are in the process of incorporating post-maintenance testing requirements for Reactor Trip System Components into plant procedures. We are also reviewing vendor and engineering recommendations to ensure appropriate test guidance is included in our test and maintenance procedures. Additionally, we are reviewing our existing technical specifications to determine if any reactor trip system post-maintenance test requirements could possibly degrade rather than enhance safety.

A statement concerning our completion of items 3.1.1 and 3.1.2 will be submitted March 1, 1984 along with a request for any necessary technical specification changes as required by item 3.1.3.

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

The present plant procedures require post-maintenance testing of all identified safety-related equipment to assure that the equipment is capable of performing its safety functions before being returned to service. Post-Maintenance testing is included in test and maintenance procedures. The administrative procedures covering control of maintenance requires that the post-maintenance testing requirements be identified and the testing signed off by the Nuclear Operations Supervisor. In conjunction with the licensing process, the Technical Specifications have been closely reviewed by LP&L. The procedure development procedure requires procedures to be consistent with all Technical Specification limiting conditions for operation and applicable vendor-recommended operating limits. This is also reviewed as part of the Technical Review Checklist which is completed for all procedures.

Maintenance procedures are reviewed whenever a revision is written or every two years. The post-maintenance operability testing and vendor and engineering recommendations will be reviewed in conjunction with the normally scheduled procedure reviews. The reviews will be completed and documented in a submittal to the NRC by November, 1985.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

We have sought the assistance of Ebasco Services and Combustion Engineering to review all vendor-recommended reactor trip breaker modifications. By February 17, 1984, we will submit to the NRC a report that will: 1) verify that each modification has been implemented; or 2) present a schedule for implementing any remaining modifications; or 3) provide a written evaluation of the technical bases for not implementing recommended modifications.

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

4.2.1 The Technical Specifications require a functional test of the Reactor Trip Breakers monthly during operation, each startup and following maintenance. This functional test (OP-903-006) will independently test the undervoltage and shunt trip devices in accordance with Combustion Engineering Availability Data Program Info Bulletin 81-02 Supplement 1.

Preventive maintenance is performed in accordance with ME-4-155. This procedure includes inspection and cleaning, contact adjustments, mechanism adjustments, insulation testing and functional testing as recommended by the vendor technical manual, General Electric Service Advice Letter (SAL) 175 dated April 2, 1979 and the Supplement to SAL 175 dated April 15, 1983.

4.2.2, 4.2.3, 4.2.4, The CE Owners Group is considering a joint effort with the B&W Owners Group to develop a reliability assurance program for the reactor trip breakers. This program is intended to gather data on AK2 breakers from in-place surveillance and maintenance programs over a two year period to assess the reliability of the breakers and their subcomponents for the reactor trip function. This program would be in lieu of any life cycle testing of the breakers.

LP&L presently intends to participate in this effort should the CEOG adopt the program. After the next CEOG meeting scheduled for late January, LP&L will be in a position to supply a more definitive schedule and response for these items. This information will be submitted by March 1, 1984.

LP&L is also using time response data to trend breaker operation according to the guidance provided by Combustion Engineering ADP Information Bulletin 83-07 dated June 15, 1983. This program may be re-evaluated or combined with the above tasks after their programs and schedules become clear.

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

This section is not applicable to Waterford 3, which is a Combustion Engineering plant.

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

This section is not applicable to Waterford 3.

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

4.5.1 Procedure OP-903-006, "Manual Reactor Trip Test" is being revised to include the required functional testing of the diverse trip features. A statement indicating implementation of this item will be submitted by January 1, 1984.

4.5.2 The Reactor Trip System is currently designed to permit periodic on-line testing. The FSAR provides adequate information for staff review of this item. No further submittals are required.

4.5.3 The C-E Owners Group has proposed a review of the existing interval for on-line functional testing as required by Technical Specifications. LP&L presently intends to participate in this effort.

Further information and schedules will be provided by March 1, 1984, following the next meeting of the CEOG.

STATE OF LOUISIANA)
) SS
PARISH OF ORLEANS)

F. W. Cook, being duly sworn, states that he is Nuclear Support and Licensing Manager of Louisiana Power & Light Company and that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this response to Generic Letter 83-28.

K W Cook
K. W. Cook

SUBSCRIBED AND SWORN to before me, a Notary Public, in and for the Parish and State above named, this 4th day of November, 1983.

Notary Public

MY COMMISSION EXPIRES:

WITH LIFE

Before the

UNITED STATES NUCLEAR REGULATORY COMMISSION

Docket No. 50-382

In the Matter of

LOUISIANA POWER & LIGHT COMPANY

Response to Generic Letter 83-28

Louisiana Power & Light Company, Applicant in the above captioned proceeding, hereby files a response to Generic Letter 83-28, for Waterford SES Unit No. 3.

Respectfully submitted,
LOUISIANA POWER & LIGHT COMPANY

By: KW Cook
K. W. Cook
Nuclear Support and Licensing
Manager

DATE: November 4, 1983

SERIAL: LAP-83-516

NOV 07 1983

Mr. Darrell G. Eisenhut, Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
UNIT NOS. 1 AND 2
DOCKET NOS. 50-400 AND 50-401
GENERIC IMPLICATIONS OF
SALEM ATWS EVENTS

Dear Mr. Eisenhut:

Carolina Power & Light Company (CP&L) has received your letter, Generic Letter 83-28, Required Actions Based on Generic Implications of Salem ATWS Events, dated July 8, 1983. Your letter requested CP&L to furnish "the status of current conformance with the positions contained" in your letter, "and plans and schedules for any needed improvements for conformance with the positions." In April 1983, the management at the Brunswick Nuclear Project (BNP) created the Salem Response Task Force comprised of on-site CP&L and contract personnel to address the implications of the Salem event. In August 1983, a project engineer was appointed from the corporate office to coordinate the responses of all of CP&L's nuclear plants. The lessons learned by the Brunswick Task Force have been applied in developing the response for the Harris plant. Pursuant to Generic Letter 82-14, CP&L hereby transmits one original and forty copies of our response to Generic Letter 83-28.

Representatives of CP&L are participating in the Nuclear Utility Task Action Committee (NUTAC), sponsored by the Institute of Nuclear Power Operations (INPO), concerning Section 2.2.2 of your letter, Vendor Interface; and in the Westinghouse Owners' Group's (WOG) generic evaluation of the positions stated in your letter. Based on our participation in these industry groups, revisions to the schedule provided in our response may be necessary as we consider the results of these studies.

The fact that the Harris Nuclear Project (HNP) is in the construction phase accounts for some of the differences in implementation between HNP, and the BNP and the Robinson Nuclear Project (RNP).

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If you have any questions regarding our response, please do not hesitate to call a member of our Licensing staff.

Yours very truly,

A. B. Cutter
Vice President
Nuclear Engineering & Licensing

DCW/ccc (8371DCWa)
Attachment

cc: Mr. B. C. Buckley (NRC)
Mr. G. F. Maxwell (NRC-SHNPP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. Travis Payne (KUDZU)
Mr. Daniel F. Read (CHANGE/ELP)
Mr. R. P. Gruber (NCUC)
Chapel Hill Public Library
Wake County Public Library
Mr. Wells Eddleman
Dr. Phyllis Lotchin
Mr. John D. Runkle
Dr. Richard D. Wilson
Mr. G. O. Bright (ASLB)
Dr. J. H. Carpenter (ASLB)
Mr. J. L. Kelley (ASLB)

A. B. Cutter, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Lisa M. Randall
Notary (Seal)

My commission expires: 5/18/88

HARRIS NUCLEAR PROJECT

RESPONSE TO GENERIC LETTER 83-28

"REQUIRED ACTIONS BASED ON GENERIC IMPLICATIONS OF SALEM ATWS EVENTS"

NOVEMBER 4, 1983

(8434DCWccc)

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

NRC POSITION

Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely. A report describing the program for review and analysis of such unscheduled reactor shutdowns should include, as a minimum:

1. The criteria for determining the acceptability of restart.

RESPONSE:

SHNPP has under development a post-trip review procedure for implementation by the on-shift operating personnel for unscheduled shutdowns (Appendix A). This procedure includes evaluation of plant conditions prior to the trip, first out annunciator, strip chart recorders on the main control board, equipment failures or transient conditions which occurred during the transient, and a review, if available, of the computer log of the event. The general criteria for determining the acceptability of a restart will be (1) the cause of the event has been identified, (2) appropriate corrective action is taken to reduce the recurrence of a trip, and (3) engineered safeguards responded in a manner consistent with the Technical Specifications. If the cause of the trip cannot be determined by the shift staff, the Post-Trip Review Report will be forwarded to the Plant Nuclear Safety Committee (PNSC) for review and recommendations. In this case, the determination of whether a unit can be restarted is made by the General Plant Manager.

NRC POSITION

2. The responsibilities and authorities of personnel who will perform the review and analysis of these events.

RESPONSE:

The responsibilities and authorities of personnel who will perform the review and unscheduled trips are specifically delineated in the SHNPP post-trip review procedure, Section 3.0.

a. The Shift Foreman is responsible for completing the post-trip review report and for determining if the cause of the trip is known and corrected. The Shift Foreman has authority to restart if the cause of the trip has been determined to be not due to equipment or instrumentation malfunction and has been corrected.

b. The Shift Technical Advisor will assist the Shift Foreman in the completion of the post-trip review report and must concur that the cause of the trip is known and corrected prior to restart.

c. The Operating Supervisor and the Manager - Operations will review the post-trip review report. These reviews may be subsequent to restart as appropriate. The Operating Supervisor and/or the Manager - Operations, as available, will be consulted and grant approval for restart for any trip for which the cause is known to be due to malfunction of equipment or instrumentation and has been corrected.

d. The SHNPP Plant Nuclear Safety Committee (PNSC) will review and make recommendations on those unscheduled trips for which the cause is not known or has not been corrected.

e. The Plant General Manager will approve restart, based on the recommendations of the PNSC, for those unscheduled trips for which the cause has not been identified or corrected.

NRC POSITION

3. The necessary qualifications and training for the responsible personnel.

RESPONSE:

The training and qualifications for personnel at the SHNPP are addressed in FSAR sections 13.1.2 and 13.2. As a minimum, the Shift Foreman, the Operating Supervisor, and the Manager - Operations will hold NRC SRO licenses. The members of the SHNPP PNSC are identified in the draft Technical Specifications previously submitted to the NRC.

NRC POSITION

4. The sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to reconstruct the event accurately and in sufficient detail for proper understanding. (See Action 1.2)

RESPONSE:

The essential sources of information which must be evaluated during the post-trip review process are identified in the post-trip procedure under development (Attachments 6.2 through 6.5 of Appendix A). The essential sources include safety-related instrumentation displays and recorders. This instrumentation is identified and described in Section 7.0 of the FSAR. Other sources of information which may be available

include the Emergency Response Facility Information System (ERFIS), the plant computer (a subpart of ERFIS), the Safety Parameter Display System (SPDS), eyewitness accounts, and logs/records maintained at SHNPP.

The ERFIS has been the subject of submittals to the NRC in response to NUREG 0737, Supplement No. 1. A brief description of the ERFIS is presented in Section 1.2.1 below.

NRC POSITION

5. The methods and criteria for comparing the event information with known or expected plant behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function).

RESPONSE:

The post-trip review procedure includes checklists (Attachments 6.2 and 6.3 of Appendix A) to review performance of engineered safeguards, such as the safety injection system and the reactor trip system, during the trip. The performance of the engineered safeguards will be compared with the required response as delineated in the SHNPP Technical Specifications.

NRC POSITION

6. A) The criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Plant Operations Review Committee, will be consulted prior to authorizing restart) and B) guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

RESPONSE

6.A The completed forms from the post-trip review will be reviewed by at least two plant personnel with current SRO licenses. If this on-shift review cannot positively identify the cause of the trip and appropriate corrective actions, the matter will be referred to SHNPP operations management and the PNSC for evaluation prior to restart. The decision to restart will be made by the Plant General Manager with recommendations from the PNSC.

6.B The post trip review report and other applicable software data will be maintained as a permanent record of the event. Guidelines for preservation of physical evidence will be incorporated in the post-trip review procedure.

NRC POSITION

7. Items 1 through 6 above are considered to be the basis for the establishment of a systematic method to assess unscheduled reactor shutdowns. The systematic safety assessment procedures compiled from the above items which are to be used in conducting the evaluation, should be in the report.

RESPONSE

NRC positions 1 through 6 were used as references when the SHNPP Post Trip/Safeguards Review procedure was developed. Section 4 of the post trip review report requires the Shift Foreman to perform a thorough systematic evaluation of the event and the plants response.

1.2 POST-TRIP REVIEW DATA AND INFORMATION CAPABILITY

NRC POSITION

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

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

   2) Parameters monitored

   3) Time discrimination between events

   4) Format for displaying data and information

   5) Capability for retention of data and information

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

RESPONSE:

1. The sequence of events during a trip can be determined using the "first out" panel for the reactor trip system; strip chart recorders in the main control room; and the plant computer, if available.

   The plant parameters which are recorded on strip charts are indicated in Section 7.5 of the SHNPP FSAR. Refer to the following tables for specific information on the noted systems:

   Table 7.5.1-1 - Containment Spray and Cooling System
   Table 7.5.1-2 - Auxiliary Feedwater System
   Table 7.5.1-3 - Control Room Emergency Filtration System

Table 7.5.1-6 - RAB Emergency Exhaust System
Table 7.5.1-9 - Reactor Coolant System
Table 7.5.1-10 - Containment System
Table 7.5.1-11 - Main Steam System
Table 7.5.1-12 - Refueling Water Storage Tank
Table 7.5.1-13 - Component Cooling Water System
Table 7.5.1-14 - Nuclear Instrumentation

The plant computer is an integral part of the SHNPP Emergency Response Information System. The ERFIS system has been described in CP&L's response to NUREG-0737 Supplement 1. The plant computer has the capability of monitoring 1500 inputs. The list of specific parameters to be recorded has not been finalized, but will be available by June 1984. The time discrimination between events averages one milli-second; the maximum discrimination is two milli-seconds. The output of the plant computer can be displayed on control room CRT's or printers; a second CRT and printer is provided as a backup in the control room. Data for each input is stored for twelve hours. This data can be manually transferred to magnetic tape. The power sources for the computer consist of non-class IE AC power and a 30-minute battery powered backup.

NRC POSITION

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.

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

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

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

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

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

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

RESPONSE:

Refer to the response to item 1.2.1.

NRC POSITION

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

RESPONSE:

Refer to discussion under 1.1.6.B and 1.2.1.

NRC POSITION

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

RESPONSE:

The systems and hardware identified above are being designed to reflect NRC's post-TMI recommendations on such systems. Additions to these systems are not contemplated at the present time.

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

NRC POSITION

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

RESPONSE:

A. Confirming Components as Safety-Related

To aid operators and craft personnel to more easily identify safety related components, SHNPP is in the process of compiling the current list of safety-related components into a definitive Q-List document. Existing Ebasco (A/E) and Westinghouse (NSSS) design documents and drawings list this information. As the Q-List is completed, the data will be reviewed to reconfirm that all components whose functioning is required to trip the reactor are identified as safety-related. This is scheduled to be complete by December, 1984. The Q-List will be used by operations and maintenance personnel while preparing final plant procedures and other safety-related activities.

B. Documenting Safety-Related Status

Q-List procedures and control instruction will be contained in the Plant Operating Manual. Plant procedures will ensure components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance work orders and parts replacement, prior to plant operation.

As the Q-List identified in A above is prepared listing safety-related components, this information will be entered into the SHNPP Equipment Data Base System (EDBS), which serves as the computerized listing of all plant equipment. Appendix B gives an overview of the data elements contained in this system. The EDBS will be used as the data base for ordering spare parts, providing schedule input for maintenance testing activities, providing input to maintenance work requests for use by the planner/analyst, and providing the material history records. Data entry is strictly controlled by appropriate plant procedures to preclude nonapproved changes or entry. When components are identified as safety-related, this information will be automatically available when work functions are planned.

This centralized system therefore provides a summary of each component's quality related designations, reference listing of related documentation such as procedures or technical manuals, spare parts information, and material history records, thus closely integrating all plant activities associated with that item. The format of the EDBS system has been developed and implemented at SHNPP, with data entry in progress. The system will be fully operational by June, 1985.

C. Vendor Interface

The Westinghouse Electric Corporation designed and provided the major Reactor Protection System components at SHNPP. Westinghouse currently utilizes a Technical Bulletins System that documents recommended changes in equipment and procedures. These bulletins also provide information concerning unique operating conditions and experience at other PWR plants. All distributions of Westinghouse safety-related Technical Bulletins are now accompanied by a return receipt. The return receipts are pre-addressed to Westinghouse for recording Bulletins transmitted and their status. Technical Bulletins for which receipt is not acknowledged within a reasonable time are retransmitted. On a periodic basis, a list of current Technical Bulletins is issued by Westinghouse. This provides a positive feedback mechanism to ensure applicable Westinghouse technical information has been received.

SHNPP is currently performing a review to verify that all Westinghouse Technical Bulletins issued to SHNPP have been received and implemented as appropriate. This review is currently scheduled to be completed by June 1985. This process ensures that the Westinghouse technical information is appropriately reviewed and incorporated, as applicable, in the plant instructions and procedures prior to and following initial plant operation.

The vendor interface program and plant procedures ensure that adequate controls are established for the Reactor Protection System components in determining that the vendor information is complete, current, and controlled prior to plant operation and throughout the life of the plant, and appropriately referenced and incorporated into the plant instructions and procedures.

For safety-related equipment requiring environmental qualification, a program is in progress to systematically review vendor data packages to identify unique features that need to be incorporated into the spare parts and maintenance activities. This will be completed by December, 1984.

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

NRC POSITION

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

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

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

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

    3) A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10CFR50, Appendix B, apply to safety-related components.

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

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*Safety-related structures, systems, and components are 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 off-site exposures comparable to the guidelines of 10CFR Part 100.

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

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

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

RESPONSE:

1. Equipment classification for safety-related equipment has been accomplished using the guidelines established in ANSI 18.2 for fluid system components and IEEE standards for electrical components. This listing by system is provided in Table 3.2.1-1 of the FSAR. Additional documents

provide the reference to the component level; including the valve list, line list, and instrument indices.

As described in our response to 2.1, SHNPP is preparing a computerized listing of all plant equipment, both safety-related and nonsafety-related to better aid the operators and craft personnel.

The data base provides sufficient information for all personnel to understand and apply the proper safety designations to all equipment in maintenance and operations related activities. It will describe if the equipment is safety-related, what procurement requirements apply to individual parts, whether the equipment is environmentally qualified, which documents such as technical manuals need to be referenced, and provides the storage of its material history. Data entry and manipulation is controlled by plant procedure 0-TMM-101 which will require appropriate levels of approval prior to data entry.

Safety-related equipment requiring environmental qualification has been tested during initial purchase by the A/E, as described in FSAR Section 3.10 and 3.11. The documentation for the test results is being reviewed to verify that the suitability of the equipment for its expected environment has been demonstrated and that unique aspects necessary to be incorporated into appropriate maintenance procedures have been identified. This will be accomplished by June, 1985.

The EDBS has been established to allow items to be also classified as "important to safety." SHNPP has not, however, committed to implementing a program for important to safety items. The results of the AIF working group addressing this issue are being followed; the results of which will be used as a basis for establishing a SHNPP approach.

2. Currently, SHNPP is phasing in a vendor interface program to provide adequate assurance that vendor information for safety-related components, which is significant to safety, is appropriately incorporated in the plant instructions and procedures. This vendor interface program will include:

1) Procedural processing of vendor recommendations

Processing of vendor recommendations will be in accordance with SHNPP procedures. Vendor recommendations concerning plant equipment will be forwarded to plant engineering for evaluation and implementation, as appropriate, in plant instructions and procedures. The vendor recommendations will also be forwarded to

the On-Site Nuclear Safety Committee
and included in the Operating Experience Feedback
Program, if appropriate.

2) Procedural control of vendor technical manuals

SHNPP maintenance activities will be performed in accordance with approved plant procedures. The vendor technical manuals are used as a source of reference material in preparing these procedures. This procedural control provides for review and incorporation of technical manual information into the plant maintenance procedures. Thus, the vendor input is evaluated before use and integrated with site specific conditions and experience. In some cases where the equipment is complex and the vendor manuals are suitable for direct use, the particular section of the manual that provides instructions for accomplishing the desired activity will be referenced or incorporated as part of the approved procedure. In such cases, the SHNPP procedure will require that the technical manual be plant approved.

SHNPP is in the process of developing a procedure for the review of safety-related vendor technical manuals in the preparation of the plant operating and maintenance instructions and procedures. This procedure is scheduled to be completed by March, 1984.

3) Control of the vendors manuals

This will be accomplished by incorporating the vendor technical manuals in the document control process as controlled documents.

The vendor interface program with the SHNPP Nuclear Steam Supply System (NSSS) vendor is described in response to 2.1.

SHNPP additionally utilizes vendor information from other industry sources, such as INPO NPRDS and Notepad, and NOMIS. Significant industry-wide events are identified and recommended action provided in INPO Significant Operating Reports (SOERs). Significant events (without recommendations) are provided in INPO Significant Event Reports.

Additionally, the NRC issues notification of safety-related concerns and regulatory requirements. IE Bulletins, Notices, and Circulars provide current information concerning component or design discrepancies. These notifications are further supported by the provision of Title 10, Chapter 1, CFR Part 21, which requires "that the directors and responsible

officers of organizations that construct, own, operate, or supply components of a facility or activity that is licensed or otherwise regulated by the Nuclear Regulatory Commission inform the Commission if they obtain information reasonably indicating that such facility, activity, or basic component fails to comply with regulatory requirements relating to substantial safety hazards or that such facility, activity, or basic component contains a defect which could create a substantial safety hazard." This effectively provides an additional safeguard for early identification of safety-related component failure or design discrepancy.

Plant surveillance testing, equipment repair and replacement procedures, and the quality assurance programs additionally provide assurance of safety-related equipment reliability.

CP&L is supporting the INPO Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2. NUTAC is currently formulating the recommendations for an industry-wide vendor information program for safety-related equipment. CP&L believes this program will provide a practical industry-wide approach to assuring safety-related equipment reliability.

In summary, the SHNPP vendor interface program in conjunction with plant procedures, surveillance testing, equipment repair and replacement, and the quality assurance program provides adequate assurance of safety-related component reliability.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

NRC POSITION

The following actions are applicable to post-maintenance testing:

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

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

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval. (Note that action 4.5 discusses on-line system functional testing.)

RESPONSE:

Verification that SHNPP post-maintenance operability testing of safety-related components in the reactor trip systems is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety function before being returned to service will be performed during development of the procedures. Applicable vendor and engineering recommendations will be incorporated during development of the procedures. This will be completed by June, 1985.

Westinghouse, under contract to the Westinghouse Owners Group, is conducting a compilation of all existing maintenance information regarding Westinghouse switchgear, including lessons learned in the post-Salem interval. This effort will be completed by the end of 1983. Any new or improved maintenance requirements resulting from cyclic test programs, such as from WOG Program "Cyclic Life and Class IE Qualification of the DB and DS Circuit Breaker Shunt Trip Attachments for Reactor Trip Switchgear; Cycle Life Testing of DS-UVTA" will be reviewed and incorporated into appropriate SHNPP maintenance procedures.

The post-maintenance test procedures will be reviewed to demonstrate that they will not degrade the performance of the equipment and its safety functions before being returned to service. Testing requirements will be incorporated into the Technical Specifications. This will be completed by June, 1985.

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

NRC POSITION

The following actions are applicable to post-maintenance testing:

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

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure

that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

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

RESPONSE:

While developing the maintenance procedures for safety-related equipment, SHNPP shall incorporate a review to ensure that post-maintenance operability testing is required to be conducted and that testing demonstrates the equipment is capable of performing its safety function prior to being returned to service. Vendor recommendations from technical manuals and technical bulletins, plus any technical specification requirements, will be incorporated during the development of plant procedures. Any work performed on safety-related equipment will be done using approved plant procedures. Portions of technical manuals may be excerpted or referenced at the time a procedure is approved. Any standardized technical specifications that are perceived to degrade rather than enhance safety will be identified during the submission of SHNPP specific technical specifications and justification provided for any deviations. Use of various surveillance test procedures will provide a documented basis of test results. This program will be implemented by June, 1985.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

NRC POSITION

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

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

RESPONSE:

The March 31, 1983, letter refers to NS-EPR-2744 E. P. Rahe of Westinghouse to R. C. DeYoung of NRC. This letter applies to DS-416 UVTA's only, and addresses UVTA dimensional variations. Subsequent to this letter, on April 21, 1983, a letter requiring

DS-416 UVTA replacement with modified shaft widened grooves for the retaining ring was issued. Refer to NS-EPR-2753, E. P. Rahe of Westinghouse to R. C. DeYoung of NRC. As indicated in the letter of April 21, Westinghouse has committed to its utilities to replace UVTAs on DS-416 reactor trip switchgear supplied by Westinghouse for its Nuclear Steam Supply System so that, 1) the new attachments have modified (widened) grooves to accommodate the new retaining rings, 2) manufacturing drawings have been revised and quality control procedures modified so that critical design dimensions are maintained during manufacture, and 3) a field installation procedure will be provided for proper alignment and interface of the attachment with the breaker trip shaft. These replacement devices will be marked with a serial numbering system.

These modifications will be completed by December, 1984.

4.2 <u>REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)</u>

<u>NRC POSITION</u>

Licensees and applicants shall describe their preventative maintenance and surveillance program to ensure reliable reactor trip breaker operation. The program shall include the following:

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

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

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

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

<u>RESPONSE</u>:

1. Based on the latest results of the life cycle testing program, and the recommendation(s) of Westinghouse, SHNPP will implement a comprehensive planned maintenance program covering the reactor trip breakers. This will be completed by December, 1984.

2. Incorporated with the planned maintenance program will be a review of the parameters necessary to be trended. These parameters will then be included under the overall plant trending program, which will be in operation by December, 1984.

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

4. The need for periodic replacement of breakers or components will be determined based on the results of life cycle testing and incorporated into the planned maintenance system by June, 1985.

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

NRC POSITION

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

RESPONSE:

SHNPP is working closely with the Westinghouse Owners Group (WOG) in addressing the actions relating to Reactor Trip System reliability, particularly with respect to the requirement for automatic actuation of the shunt trip attachment and the on-line surveillance requirement. A detailed generic design package for incorporation of an automatic shunt trip feature into various Westinghouse Reactor Protection Systems has been developed under WOG sponsorship. The complete generic design package of the automatic shunt trip modification was submitted to NRC on June 14, 1983, J. J. Sheppard, Chairman of WOG by letter OG-101.

The generic design package of the automatic shunt trip modification contains a design basis, functional requirements, conceptual design, and assessment of conformance to safety criteria. The design of the system includes hard-wired and component installation provisions for on-line surveillance testing that independently verifies by manual means the operability of the UVTA and the automatic shunt trip.

The NRC issued a favorable Safety Evaluation Report on the generic design on August 10, 1983 (letter from D. Eisenhut to J. J. Sheppard). The SER lists plant-specific information required for individual plant modifications, which will be provided by December, 1984. SHNPP has committed to installing a class IE automatic shunt trip attachment.

4.4 N/A TO SHNPP

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

NRC POSITION

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants.

1. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W (see Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

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

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

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

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

RESPONSE:

1. As noted above in Section 4.3, the generic design package for the automatic shunt trip furnished to WOG for submittal to the NRC included an installation for on-line surveillance testing of the UVTA and automatic shunt trip that provided independent verification of each attachment. The existing generic RTS automatic trip is by UVTA only and

manual trip by either UVTA or shunt trip. Although the existing generic system does not include installed provisions for independent verification of the two trips, a procedure for performing independent verification during shutdown is feasible and has been recommended by Westinghouse in Westinghouse Technical Bulletin NSD-TB-83-03, dated March 24, 1983. The procedure is intended only to provide general guidance from which SHNPP can develop its own plant-specific procedure. This bulletin addresses the concerns discussed in NRC IE Circular 81-12 and I.E. Bulletin 83-01. It is noted that the procedure presented in this Tech Bulletin is not performed at power, since actuation of a manual trip on the generic Westinghouse reactor trip system would trip the reactor.

2. SHNPP will have on-line testing capability.

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

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

APPENDIX A

APR008

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

UNIT 0

PLANT OPERATING MANUAL

VOLUME 3

PART 1

PROCEDURE TYPE: OPERATIONS MANAGEMENT MANUAL (OMM)

NUMBER: 0-OMM-04

TITLE: POST TRIP/SAFEGUARDS REVIEW

REVISION 0

APPROVED: ______________________________ ____________________
Signature Date

TITLE: ______________________________

(DRAFT)

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1.0 PURPOSE

1.1 The purpose of this procedure is to establish the requirements to perform and the methodology for performing a formal post event review for all reactor trips or safeguards actuations. This procedure establishes the requirements and criteria that must be met prior to the start-up of the unit following a reactor trip or safeguards event.

2.0 REFERENCES

2.1 NUREG-1000 Generic Implications of ATWS Event at Salem Nuclear Power Plant - Volume 1, Section 1.1

2.2 Procedures Administration Manual

3.0 RESPONSIBILITIES

3.1 Shift Foreman

3.1.1 Ensure the Post Trip/Safeguards Review Report is complete.

3.1.2 Provide central direction for the investigation of the event.

3.1.3 Authorizes restart if the cause of the trip has been determined to be not due to equipment or instrumentation malfunction and has been corrected.

3.1.4 May authorize the withdrawal of the shutdown banks provided his estimated critical position is avoided by at least 1000 PCM, as determined by the use of current approved rod worth and boron worth curves.

3.2 Shift Technical Advisor

3.2.1 Assist and advise the Shift Foreman in the completion of the Post Trip/Safeguards Review Report and must concur that the cause of the trip is known and corrected prior to restart.

3.3 Operating Supervisor (or Designated Alternate)

3.3.1 Reviews the Post Trip/Safeguards Review Report.

3.3.2 Authorizes restart, if the Manager - Operations is unavailable, for any trip for which the cause is known to be due to malfunction of equipment or instrumentation and has been corrected.

3.4 Manager - Operations (or Designated Alternate)

3.4.1 Reviews Post Trip/Safeguards Review Report.

3.0 RESPONSIBILITIES (Cont'd)

3.4.2 Authorizes restart, for any trip for which the cause is known to be due to malfunction of equipment or instrumentation and has been corrected.

3.5 Plant General Manager (or Designated Alternate)

3.5.1 Reviews Post Trip/Safeguards Review Report.

3.5.2 Approves restart, based on the PNSC recommendations, for those trips for which the cause has not been identified or corrected.

3.6 Plant Nuclear Safety Committee (PNSC)

3.6.1 Reviews the Post Trip/Safeguards Review Report.

3.6.2 Reviews the Post Trip/Safeguards Review Report prior to restart, if applicable.

3.6.3 Recommends restart, for those trips for which the cause has not been identified or corrected.

3.7 PNSC Chairman

3.7.1 Approves the Post Trip/Safeguards Review Report.

3.7.2 Recommends restart, if appropriate.

4.0 DEFINITIONS/ABBREVIATIONS

4.1 Definitions

4.1.1 Reactor Trip - a reactor trip occurs anytime the reactor trip breakers open that causes one or more full length control rods to be inserted, except planned trips during performance of an approved procedure.

4.1.2 Safeguards Actuation - any manual or automatic safeguards signal that actuates or should have actuated the safeguards sequencers, except planned actuations during performance of an approved procedure.

4.2 Abbreviations

4.2.1 PNSC - Plant Nuclear Safety Committee

5.0 PROCEDURE - POST TRIP/SAFEGUARDS REVIEW REPORT

The Post Trip/Safeguards Review Report (Attachment 6.1) will be completed by an individual or team qualified to assess the event. Individuals qualified to assess the event include licensed personnel, qualified Shift

5.0 PROCEDURE - POST TRIP/SAFEGUARDS REVIEW REPORT (Cont'd)

Technical Advisors or other experienced personnel assigned by Plant Management. The Post Trip/Safeguards Review Report does not take priority over any actions required to place the Plant in a safe condition.

5.1 Data Collection

The purpose of the Data Collection section of the form is to obtain data as quickly as possible to evaluate the event.

Table 1, Attachment 6.2, is for verification of certain automatic reactor trip actions.

Table 2, Attachment 6.3, is to be completed only in the event a safeguards actuation occurs.

Table 3, Attachment 6.4, is for recording strip chart data. Data to be recorded includes the value of the variable immediately prior to the event, the maximum and minimum value of the variable during the event, and the value of the variable after the event has stabilized. Each strip chart is to be labeled with the start and stop time of the event, date, and initials. Additional strip chart data should be added to the form if the variable is relevant to the event.

Table 4, Attachment 6.5, is to be completed only if the sequence of events printout from the computer is available. Table 4 is used to verify the proper operation of the reactor trip breakers using the sequence of events data.

Event summaries are to be obtained from individuals who were present when the event occurred and who were actively involved in the cause and mitigation of the event. Each individual involved in the event may prepare a written summary of his involvement in the event, sign it and forward it to the Shift Foreman. Also, the individuals involved may as a group or groups discuss the event. The individual leading the discussion will be responsible for preparing a summary of the discussion, signing it, and forwarding it to the Shift Foreman.

All event summaries should be obtained prior to any individual, who was involved in the event, leaving the Plant site.

Any other physical components that may be relevant to the trip event should be forwarded with the post trip review report if feasible (i.e., blown fuses, small failed components, photographs of larger components if possible).

5.2 Initial Plant Conditions

The intent of this section is to obtain initial Plant conditions immediately prior to the event.

The sum of the shutdown bank trip and the control bank trip equal the total reactor trip number.

5.3 Cause of Trip/Safeguards Actuation

This section provides for a concise summary of the event summaries obtained. This section is completed without the benefit of the sequence of events printout in order to obtain a single concise summary of the event based on observations.

5.4 Evaluation of Trip/Safeguards Actuation

The purpose of this section is to analyze the information obtained from data collections to ensure the reactor protection/safeguards equipment and first out annunciator panel operated properly.

5.4.1 Sequence of Events - This section provides for the comparison of the first annunciator and sequence of events printout, if available, to identify any discrepancies.

5.4.2 Additional First Out Annunciators - This section provides for the evaluation of the proper operation of the First Out Annunciator Panel and for the identification of other potential equipment problems.

5.4.3 Strip Chart Data - This section provides for the evaluation of trend data from Table 3, Attachment 6.4, against expected responses based on operator experience and training.

5.4.4 Detail Summary of Event - This section provides for the detailed summary of the event based upon all of the available information.

5.4.5 Cause of Event - This section provides for establishing the cause of the event and written justification.

5.4.6 Malfunctions - This section provides for establishing the cause of any protection/safeguards equipment malfunctions or first out annunciator malfunctions.

5.4.7 Other - This section provides for listing other equipment malfunctions which, if it had functioned properly, would have helped mitigate the event.

5.5 Corrective Actions

This section provides for the documentation of the corrective action taken as a result of the event. Any long term improvements identified should be submitted to Plant management utilizing the Plant Improvement Form.

5.6 Signatures

This section provides a list of all individuals involved with the Post Trip/Safeguards Review Report and review by two current SRO licensed individuals.

5.7 Post Trip/Safeguards Start-up Authorization

The Shift Foreman may authorize restart of the unit if the cause of the event has been clearly identified and corrected and was not due to equipment or instrumentation malfunction(s). The approval of the Operating Supervisor or Manager - Operations is required if the cause of the event has been clearly identified and corrected and was due to equipment or instrumentation malfunction(s). If the cause of the event has not been clearly identified or there are questions concerning the proper performance of protection/safeguards equipment or systems during the event, the Post Trip/Safeguards Review Report will be reviewed by the PNSC. The PNSC shall make recommendations on the restart of the unit. Upon completion of any additional corrective actions and the review thereof, the PNSC Chairman may recommend the restart of the Unit. The Plant General Manager may authorize the restart of the unit based on the PNSC's recommendations.

This section also provides for the closeout of the Post Trip/Safeguards Review Report process if a restart is not planned.

5.8 Scheduled Review

This section defines the normal review of the Post Trip/-Safeguards Review Report.

6.0 ATTACHMENTS

6.1 Post Trip/Safeguards Review Report

6.2 Table 1 - Reactor Trip Automatic Action Verification

6.3 Table 2 - Safeguards Automatic Action Verification

6.4 Table 3 - Strip Chart Data

6.5 Table 4 - Sequence of Events Summary

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
POST TRIP/SAFEGUARDS REVIEW REPORT

1.0 DATA COLLECTION

1.1 Complete Attachment 6.2, Table 1, Reactor Trip Automatic Actions Verification.

1.2 Complete Attachment 6.3, Table 2, Safeguards Automatic Actions Verification, if applicable.

1.3 Complete Attachment 6.4, Table 3, Strip Chart Data.

1.4 Complete Attachment 6.5, Table 4, Sequence of Events Summary, if sequence of events printout is available.

1.5 Attach event summaries from people involved in the event or a summary of discussion with people involved in the event.

2.0 INITIAL PLANT CONDITIONS

2.1 Date of Event ______________ Time of Event __________

Shutdown Bank Trip No.: ______________
Control Bank Trip No.: ______________
Total Reactor Trip No.: ______________

2.2 Personnel on Duty

Shift Foreman: ______________
Senior Control Operator: ______________
Control Operators: ______________
____
____
Shift Technical Advisor: ______________

2.3 Plant Conditions

2.3.1 Reactor Power ______% TAVE ________°F Net MWe ______

2.3 Plant Conditions (Cont'd)

2.3.2 List evolutions in progress on production or LCO equipment immediately prior to the event (i.e., surveillance testing, trouble shooting, maintenance, unit start-up activities, unit shutdown activities, or other activities which could have contributed to the event).

____
____
____
____
____

2.3.3 List any equipment that was inoperable which could have contributed to the event (i.e., major production equipment, LCO equipment, or instrumentation and controls equipment).

____
____
____
____
____

3.0 CAUSE OF TRIP/SAFEGUARDS ACTUATION

3.1 Reactor Trip Actuation

Record First Out Annunciator ______________________________

Briefly describe the cause of actuation based on review of event summaries: ______________________________
____
____
____

3.2 Safeguards Actuation

If safeguards actuated, briefly describe cause of actuation based on review of event summaries: ______________________________
____
____
____
____
____

4.0 EVALUATION OF TRIP/SAFEGUARDS ACTUATION

4.1 Sequence of Events

Does the sequence of events printout (if available) agree with the First Out Annunciator? YES_______ NO_______

If no, explain discrepancies.

____
____
____
____
____

4.2 Additional First Out Annunciators

4.2.1 Briefly describe any additional First Out Annunciators identified: ______________________________
____
____
____

4.2.2 List and explain any other First Out Annunciators which should have annunciated but did not: ______________
____
____
____
____

4.3 Strip Chart Data

List and explain any unexpected strip chart responses from Attachment 6.4, Table 3.

____
____
____
____

4.4 Detail Summary of Events

Explain in detail the events which led to the actuation of the Trip/Safeguards and the actions performed to place the Plant in a stable condition (reflect the event summaries and the sequence of events printout, if available).

____
____
____
____
____
____
____
____
____
____

4.0 EVALUATION OF TRIP/SAFEGUARDS ACTUATION (Cont'd)

4.5 Describe the cause of the event and justification:

4.6 Describe the cause of any malfunction(s) of the protection/safeguards systems or First Out Annunciators:

4.7 List equipment other than protection/safeguards equipment which failed to function properly during the event which, if it had functioned properly, would have helped mitigate the event.

5.0 CORRECTIVE ACTIONS

5.1 Describe actions taken to correct cause of the event:

5.0 CORRECTIVE ACTIONS (Cont'd)

5.2 Describe corrective actions for any protection/safeguards equipment which failed to function properly during the event: ______________________________
____
____
____
____
____

5.3 Describe any corrective actions for any First Out Annunciators which should have annunciated but did not:
____
____
____
____
____

5.4 List items identified in Step 4.7 which are repaired prior to restart: ______________________________
____
____

5.5 Describe any corrective action for equipment which will not be repaired prior to restart and justification:
____
____
____
____
____

6.0 POST TRIP/SAFEGUARDS REVIEW REPORT PREPARATION

List the individuals assisting in the preparation of the report:
____
____
____
____
____
____

Reviewed by: ____________________ Date ______ Time ______
SRO Licensed: __________ __________ Date ______ Time ______
Signature Title
__________/Shift Foreman Date ______ Time ______
Signature Title

7.0 POST TRIP/SAFEGUARDS START-UP AUTHORIZATION

7.1 I have reviewed the Post Trip/Safeguards Review Report. The cause of the event has been clearly identified, has been corrected, and was not due to equipment or instrumentation malfunction(s). All Protection/Safeguards equipment or systems functioned as designed.

Start-up Approved By:

_______________________________ Date _______ Time _______
Shift Foreman

Concurrence By:

_______________________________ Date _______ Time _______
Shift Technical Advisor

7.2 I have reviewed the Post Trip/Safeguards Review Report. The cause of the event has been clearly identified has been corrected, and was due to equipment or instrumentation malfunction(s). All Protection/Safeguards equipment or systems functioned as designed.

Start-up Approved By:

_______________________________ Date _______ Time _______
Signature

____
Operating Supervisor or
Manager - Operations

Concurrence By:

_______________________________ Date _______ Time _______
Shift Technical Advisor

7.0 POST TRIP/SAFEGUARDS START-UP AUTHORIZATION (Cont'd)

7.3 If the cause of the event has not been clearly identified, the cause of the event has not been corrected, or there are questions concerning the proper performance of protection/safeguards equipment or systems during the event, PNSC review required prior to restart.

Post Trip/Safeguards Review Report Submitted to PNSC
______________________________ Date ______ Time ______
Shift Foreman

Additional corrective actions identified by PNSC:
____
____
____
____
____

Additional corrective actions completed if necessary. PNSC has reviewed and recommends start-up.

Start-up-up Recommended By:
______________________________ Date ______ Time ______
PNSC Chairman

Start-up Approved By:
______________________________ Date ______ Time ______
Plant General Manager

8.0 SCHEDULED REVIEW

8.1 Send "For Information Only" copy of the Post Trip/-Safeguards Review Report to Regulatory Compliance.

8.2 Send "For Information Only" copy of the Post Trip/-Safeguards Review Report to ONSITE Nuclear Safety.

8.3 The Post Trip/Safeguards Review Report is to be reviewed by the PNSC at its next monthly meeting or sooner if deemed necessary by the Plant General Manager.

Reviewed By: ________________________ Date __________
Operating Supervisor

Reviewed By: ________________________ Date __________
Manager - Operations

Approved By: ________________________ Date __________
PNSC Chairman

TABLE 1
REACTOR TRIP AUTOMATIC ACTION VERIFICATION

AUTOMATIC ACTION	Automatic Function Occurred	Manual Function Required	Would Not Function	Comments
1. Reactor Trip Breaker A Trip				
2. Reactor Trip Breaker B Trip				
3. Reactor Trip Bypass Breaker A Trip				
4. Reactor Trip Bypass Breaker B Trip				
5. Turbine Trip (All Turbine Valves Shut)				
6. Feedwater Regulator Valves close when Tavg decreases to 564°F				

ANY OF THE ABOVE ITEMS MAY BE MARKED N/A IF IT WAS IN THE POST TRIP POSITION PRIOR TO THE EVENT.

1. Record First Out Annunciator: ____________________

2. Record Additional First Out Annunciators Received: ____________________
____
____
____
____

Signature ______________________________ Date __________ Time __________

TABLE 2
SAFEGUARDS AUTOMATIC ACTION VERIFICATION

SAFEGUARDS EQUIPMENT	Automatic Function Occurred	Manual Function Required	Would Not Function	Comments
1. Charging/SI Pump 1A-SA				
2. Charging/SI Pump 1B-SB				
3. Charging/SI Pump 1C-SAB				
4. RHR Pump 1A-SA				
5. RHR Pump 1B-SB				
6. Emergency Service Water Pump 1A-SA				
7. Emergency Service Water Pump 1B-SB				
8. Emergency Service Water Booster Pump 1A-SA				
9. Emergency Service Water Booster Pump 1B-SB				
10. Containment Fan AH-1				
11. Containment Fan AH-2				
12. Containment Fan AH-3				
13. Containment Fan AH-4				
14. Auxiliary Feedwater Pump A				
15. Auxiliary Feedwater Pump B				
16. Emergency DG 1A-SA				
17. Emergency DG 1B-SB				
19. C.R. Vent. Isolation				

TABLE 2
SAFEGUARDS AUTOMATIC ACTION VERIFICATION (Cont'd)

CONTAINMENT SPRAY EQUIPMENT
1. Containment Spray Pump 1A-SA
2. Containment Spray Pump 1B-SB
3. Steam Line Isolation (A,B,C MSIV's Closed)

LIST ANY NECESSARY PINK STATUS LIGHTS THAT DID NOT ILLUMINATE.

____

____

____

SIGNATURE ________________________ Date ________ Time _______

TABLE 3
STRIP CHART DATA

STRIP CHART VARIABLE	Value Immediately Prior	Max Value During Event	Min Value During Event	Value After Event Stabilized
1. S/G A Narrow Range Level LR-478				
2. S/G B Narrow Range Level LR-488				
3. S/G C Narrow Range Level LR-498				
4. Pressurizer Level LR-459				
5. Pressurizer Pressure PR-444				
6. T - Avg. TR-408				
7. Loop 1 Hot Leg Temp. TR-413				
8. Loop 2 Hot Leg Temp. TR-413				
9. Loop 3 Hot Leg Temp. TR-413				
10. Loop 1 Cold Leg Temp. TR-410				
11. Loop 2 Cold Leg Temp. TR-410				
12. Loop 3 Cold Leg Temp. TR-410				
13. Nuclear Power Range N-45				
14. S/G A Wide Range Level LR-477				
15. S/G B Wide Range Level LR-477				
16. S/G C Wide Range Level LR-477				
17. RCS Wide Range Loop A Pressure PR-402				
18. RCS Wide Range Loop B Pressure PR-402				

TABLE 3
STRIP CHART DATA (Cont'd)

Label each strip chart lested above with the start and stop time of the event, date, and initials.

SIGNATURE ______________________ Date ________ Time ________

List any unexpected strip chart responses ______________________
____
____
____

SIGNATURE ______________________ Date ________ Time ________

TABLE 4
SEQUENCE OF EVENTS SUMMARY

TIME SEQUENCE OF EVENTS STARTED: HOUR________ MINUTES________ SECONDS _____
INITIATING EVENT ____________________________________ (0 Cycles)
1st Reactor Trip Signal Initiated at ____________________ Cycles
1st Safeguards Initiating Event at ______________________ Cycles

SUBSEQUENT ALARMS	Computer Address Id	Cycle Time	Delta Time In Cycles	Acceptable Delta Time In Cycles
1. Reactor Trip Breaker A Trip				≤(Later)
2. Reactor Trip Breaker B Trip				≤(Later)
3. Reactor Trip Bypass Breaker A Trip				≤(Later)
4. Reactor Trip Bypass Breaker B Trip				≤(Later)
5. TB HYD OIL LO P (Turbine Trip)				≤(Later)
6. Reactor Manual Trip Breaker 1 Trip			N/A	N/A
7. Reactor Manual Trip Breaker 2 Trip			N/A	N/A

Any items 1-5 above may be marked N/A if it was in the Post Trip position prior to the event. Items 6 & 7 should be completed only if a manual trip button was pushed subsequent to the first trip initiating signal.

The delta time in cycles is the time between the subsequent alarm and the 1st trip initiating event.

DELTA TIME = (Subsequent Alarm Time in Cycles) - (1st Trip Initiating Event in Cycles)

If the delta time does not meet the acceptance criteria, it will be evaluated in Section 4.0 of the Post Trip/Safeguards Review Report.

NOTE: 50 Cycles = 1 Second

Attach the Sequence of Events printout and the Post Trip Review printout to Attachment 6.1.

SIGNATURE____________________________ Date _________ Time _________

APPENDIX B

APPENDIX B

0-PLP-603 EQUIPMENT DATA BASE

1.0 Purpose

The purpose of the Equipment Data Base System is to provide a consistent, centralized means of identifying each plant structure, system, and component and storing any information relating to those structures, systems, and components so that it is easily and conveniently retrievable.

2.0 References

The Equipment Data Base System in itself is not specifically required by Technical Specifications, the FSAR, or any regulation; however, the information contained therein is required by many such commitments, as is an efficient method for storage and retrieval of such information.

3.0 Responsibilities

Responsibilities for the creation, maintenance, and utilization of the Equipment Data Base System will be detailed in Section 6.0 (Later), Implementation, of this program.

4.0 Definitions

CMMS - Corporate Materials Management System
MMS - Maintenance Management System
NPRDS - Nuclear Plant Reliability Data System

5.0 General

The Equipment Data Base is an organized table of equipment-related information. Within each plant, the table is organized first by unit, then by systems within each unit, by functional equipment designations within each system, and by specific occurrence of each functional equipment type. For instance, Cooling Tower Makeup Pump 1X-NNS for unit 1 at SHNPP has a plant code of 03 for SHNPP, a unit code of 1, a system code of 4030 for the Cooling Towers Makeup System, an equipment code of CBU for Cooling Tower Makeup Pump, and a specific identifier of 1X-NNS.

The Equipment Data Base serves as the central repository of all equipment-related information utilized by various plant and corporate systems and groups. It stores all component information required by Maintenance Management, such as component identifiers, applicability of various special programs, level of review required, safety class, and spare part availability. It also can store the history of each maintenance activity, both corrective and periodic.

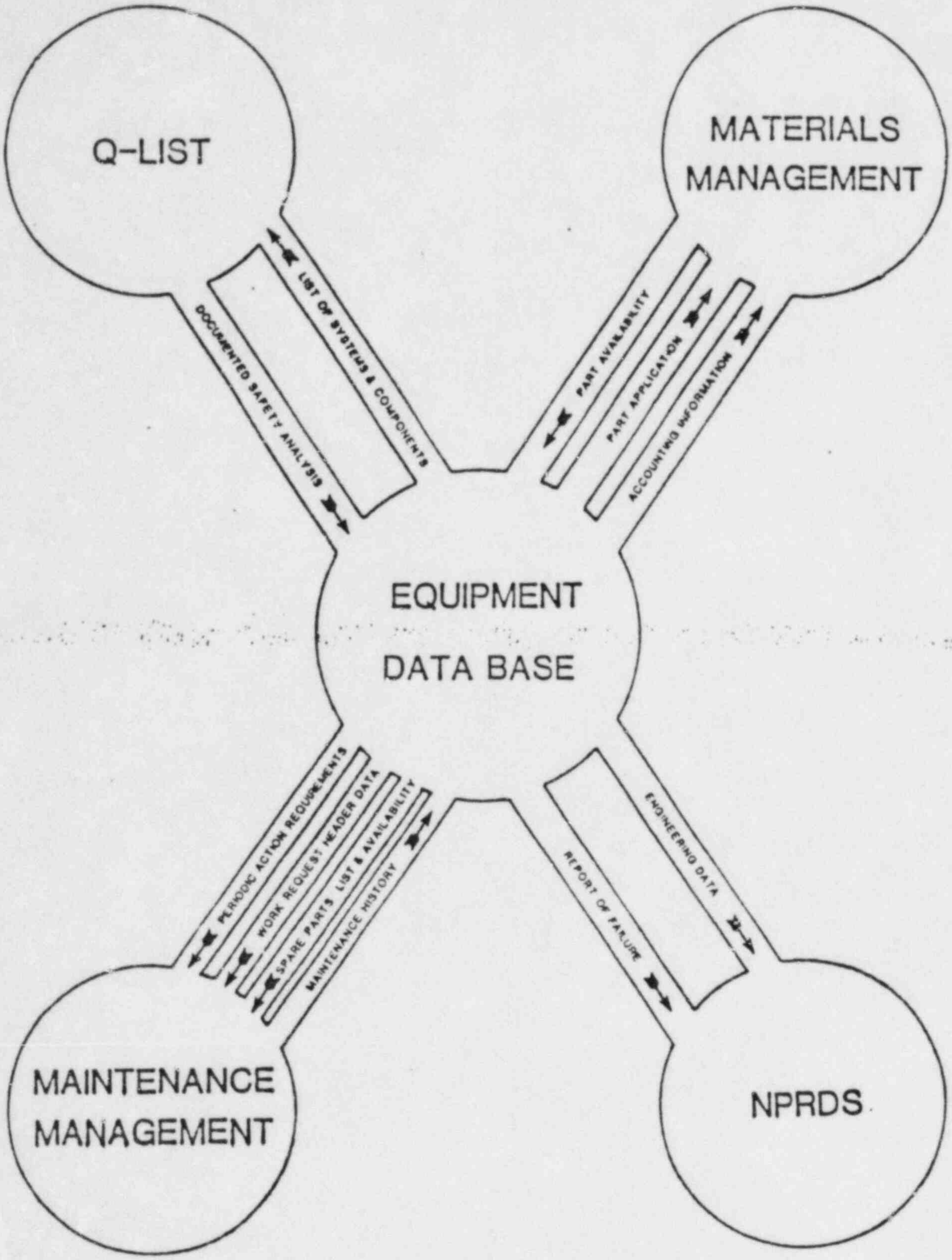
The Quality classification of a component is determined by analyzing the component against various criteria. The Equipment Data Base allows the storage and recall of this analysis, and the analysis of each repair part related to the component.

Carolina Power and Light Company is committed to support the Nuclear Plant Reliability Data System (NPRDS). This system maintains a file of engineering and failure history data for selected safety-related components within a plant. The Equipment Data Base serves as the entry point and storage location for that information at CP&L.

Safety-related electrical components installed in a harsh environment must demonstrate their qualification to function in that environment. The Equipment Data Base provides storage for both the requirements and the demonstrated capabilities of these components.

Through these and other types of information, the Equipment Data Base serves as the common tie between many separate functions. This is illustrated by attachment 1.

Copies of some of the displays and brief descriptions of the data displayed are found in attachment 2.



ATTACHMENT 1

Data Descriptions

1 - Component identification
2 - Accounting (Charge #) information
3 - Work Request Header information
4 - Pending actions
5 - Equipment out-of-service data
6 - LCO status
7 - Physical location within the plant
8 - Regulatory references
9 - Drawings which reference this system or component
10 - Technical Manuals which reference this system or component
11 - Specifications which reference this system or component
12 - Procedures which reference this system or component
13 - Special tool or equipment notes
14 - Design Basis classifications
15 - In-Service-Inspection applicability
16 - Environmental Qualification applicability
17 - NPRDS reportability
18 - Importance to fire protection
19 - Criticality to power production capability
20 - Energy Industry Identification System classifications
21 - NPRDS identification codes
22 - In-service date
23 - Applicable code or standard
24 - Manufacturer and vendor identification
25 - Manufacturer's model and serial number
26 - Manufacturer or vendor's component identifiers
27 - Component physical and operating parameters
28 - NPRDS operating data codes
29 - Quality classification justification
30 - Quality class
31 - Review and approval documentation
32 - Periodic Action Procedure identification
33 - Periodic Action performance responsibility
34 - Plant mode required for periodic action performance
35 - Estimated time required
36 - Periodicity, including tolerance
37 - Origin of commitment to perform this particular periodic action
38 - Last date periodic action was performed for this component
39 - Range of dates for next required performance
40 - Work request #
41 - Time of failure
42 - Plant status at time of failure
43 - Failure description narrative
44 - Failure cause narrative
45 - Corrective action narrative
46 - Failure codes for trend analysis
47 - Licensee Event Report # and date
48 - Normal environment at component's location
49 - Environmental parameters to which component must be qualified
50 - Identification of actual component installed
51 - Demonstrated capability of actual component installed

EQUIPMENT DATA BASE *** GENERAL INFORMATION

PLANT	UNIT	SYSTEM	EQUIPMENT	SPECIFIC ID	TAG #
03	1	4065	EZK	SW-B1SA	1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE 1

CHARGE TO H31 CAPITAL 321 MAINT 529 OPERATIONS 524 TAX CODE WA 2

Q-CLS VR ISI FIRE LCO RQD SECURITY ENVIR SEIS NPRD CRIT RWP RQD OWP

N 3

****

ACTIONS PENDING: 4

TYPE # DESCRIPTION EST HR

MORE N

****

OUT OF SERVICE DATA: 5

TIME OUT TIME DUE BACK DOCUMENTATION

CLEAR Y/N

LCO # LCO EXPIRATION 6

FUNCTION PLANT UNIT SYSTEM EQUIP SPEC-ID

CPL #/INDEX TYPE

EQUIPMENT DATA BASE *** ENVIRONMENTAL QUALIFICATION DATA

PLANT UNIT SYSTEM EQUIPMENT SPECIFIC ID TAG #

03 1 4065 EZK SW-B1SA 1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE 1

****

NORMAL ENV: TEMP DEG F PRESS PSIA REL HUM 48 % RAD MR/HR

****

QUALIFICATION REQUIREMENTS: 49

TEMP DEG F PRESS PSIA REL HUM % AGING YR SUBMERGE

CHEM SPRAY CONTENT RATE GPM DURATN MIN

CUM. RADIATION (RADS) 2HR 30DAYS 1YR 40YRS

DATA SOURCE:

****

QUALIFICATION OF INSTALLED EQUIPMENT:

MANU MOD S/N 50

TEMP DEG F PRESS PSIA REL HUM % AGING YR SUBMERGE

CHEM SPRAY CONTENT RATE GPM DURATN MIN

51

CUM. RADIATION (RADS) 2HR 30DAYS 1YR 40YRS

DATA SOURCE:

FUNCTION 208 PLANT 03 UNIT 1 SYSTEM 4065 EQUIP EZK SPEC-ID SW-B1SA

CPL #/INDEX TYPE

MSG

EQUIPMENT DATA BASE *** REFERENCE DATA

PLANT	UNIT	SYSTEM	EQUIPMENT	SPECIFIC ID	TAG#
03	1	4065	EZK	SW-B1SA	1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE (1)

****

LOCATION (7)

REG REF (8) MORE N NO

DRAWINGS (9) MORE N NO

TECH MANUALS (10) MORE N NO

SPECIFICATIONS (11) MORE N NO

****

PROCEDURES TYPE # REV DESCRIPTION

(12)

MORE N NO

****

(13)

SPECIAL TOOLS OR EQUIPMENT

MORE N NO

FUNCTION 202 PLANT 03 UNIT 1 SYSTEM 4065 EQUIP EZK SPEC-ID SW-B1SA

CPL #/INDEX TYPE

MSG

EQUIPMENT DATABASE***ENGINEERING DATA

PLANT UNIT SYSTEM EQUIPMENT SPECIFIC ID TAG #

03 1 4065 EZK SW-B1SA 1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE 1

Q CLS- N Q GRP- SAFE CLS- 14 SEIS- ISI- 15 ENV- 16 NPRD- 17 FIRE- 18 CRIT- 19

EIIS SYSTEM COMPONENT FUNCTION 20 COMPONENT APPLICATION

NPRD: SYS COMP MODE ENVIRONMENT: INT 21 EXT

IN SERVICE DATE / 22 / APPLICABLE CODE OR STD 23

MANUF: CPL- NPRD-
24
VENDOR: CPL- NPRD-

MDL# SER# 25

VENDOR COMP/SYS ID 26

DATA CD DESC CD DESC CD DESC

A B C 27

D E F

G DSC H DSC

J DSC

OPS DATA OUT-OF-SERVICE- TEST TYPE FREQ INTERVAL HOURS

CRIT HOURS FOR-TESTING DATA: CHECK

S-BY HOURS FUNCT 28

S-DN HOURS CALIB

FUNCTION 203 PLANT 03 UNIT 1 SYSTEM 4065 EQUIP EZK SPEC-ID SW-B1SA

CPL #/INDEX TYPE

EQUIPMENT DATABASE *** QUALITY CLASSIFICATION ANALYSIS

PLANT	UNIT	SYSTEM	EQUIPMENT	SPECIFIC ID	TAG #
03	1	4065	EZK	SW-B1SA	1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE

1

AFFECTED CRITERIA:

CLASS-# DESCRIPTION OF INFLUENCE A/P

-

-

-

- 29

-

-

MORE Y/N N NO *** ACCIDENT IMPACT ***

- - - - - - - - - - -

- - - - - - - - - MORE Y/N N NO

RESULTANT QUALITY CLASS _ 30

ANALYST:	TITLE:	DATE:
REVIEWED: 31	TITLE:	DATE:
APPROVED:	TITLE:	DATE:

FUNCTION 204 PLANT 03 UNIT 1 SYSTEM 4065 EQUIP EZK SPEC-ID SW-B1SA

CPL #/INDEX TYPE

MSG

EQUIPMENT DATA BASE *** PERIODIC ACTION REQUIREMENTS

PLANT UNIT SYSTEM EQUIPMENT SPECIFIC ID TAG #

03 1 4065 EZK SW-B1SA 1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE 1

****

TYPE # DESCRIPTION 32 RESP 33 MOD 34 EST HR 35 FREQ/ TOL /UNITS 36

COMMITMENT ORIGIN 37 LAST ACTION DATE 38 EARLY NEXT LATE NEXT 39

****

32 33 34 35 + - 36

37 38 39

****

32 33 34 35 + - 36

37 38 39

****

32 33 34 35 + - 36

37 38 39

****

MORE Y/N N

FUNCTION 205 PLANT 03 UNIT 1 SYSTEM 4065 EQUIP EZK SPEC-ID SW-B1SA

CPL #/INDEX TYPE

MSG ENTER ADDITIONAL DATA

EQUIPMENT DATABASE *** FAILURE HISTORY

PLANT	UNIT	SYSTEM	EQUIPMENT	SPECIFIC ID	TAG #
03	1	4065	EZK	SW-B1SA	1SW-1

VALVE, ISOLATION

DESCRIPTION ESW HDR A AUX RES INTAKE

1

****

EVENT DATA: CONTROL # EVENT START EVENT END STATUS AT TIME OF FAILURE

40 : 41 : 42

FAILURE DESCRIPTION:

43

MORE Y/N

CAUSE OF FAILURE:

44

MORE Y/N

CORRECTIVE ACTION:

45

MORE Y/N

******* PRIOR EVENT Y/N *** ****** ADD EVENT Y/N **************************

FAILURE ANALYSIS: TYPE MODE CAUSE

EFFECT DETECTION 46 ACTION TAKEN

LER # 47

LER SUBMITTED

FUNCTION 207 PLANT 03 UNIT 1 SYSTEM 4065 EQUIP EZK SPEC-ID SW-B1SA

CPL #/INDEX TYPE

MSG

PACIFIC GAS AND ELECTRIC COMPANY

PG&E + 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

J. O. SCHUYLER
VICE PRESIDENT
NUCLEAR POWER GENERATION

November 7, 1983

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

Re: Docket No. 50-275, OL-DPR-76
Docket No. 50-323
Diablo Canyon Units 1 and 2
Generic Letter No. 83-28 - Required Actions
Based on Generic Implications of ATWS Events

Dear Mr.Eisenhut:

Enclosed is the status of current conformance and the schedules for improvements planned for Diablo Canyon Units 1 and 2 to assure compliance with Generic Letter 83-28, "Required Actions Based on Generic Implications of ATWS Events".

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Subscribed to in San Francisco, California this 7th day of November, 1983.

Respectfully submitted,

Pacific Gas and Electric Company

By J. O. Schuyler
J. O. Schuyler
Vice President
Nuclear Power Generation

Robert Ohlbach
Philip A. Crane, Jr.
Richard F. Locke
Attorneys for Pacific
Gas and Electric Company

By Malcolm H. Furbush
Malcolm H. Furbush

Subscribed and sworn to before me
this 7th day of November, 1983.

Nancy J. Lemaster
Nancy J. Lemaster, Notary Public in
and for the City and County of
San Francisco, State of California

SEAL

My commission expires April 14, 1986.

NANCY J. LEMASTER
NOTARY PUBLIC-CALIFORNIA
CITY AND COUNTY OF
SAN FRANCISCO
My Commission Expires April 14, 1986

Enclosure

cc: J. B. Martin, NRC
Service List

A003
1/1

ENCLOSURE

RESPONSE TO GENERIC LETTER 83-28
STATUS OF CURRENT CONFORMANCE, PLANS, AND SCHEDULES

This enclosure describes the status of current conformance and the schedules for improvements to assure compliance of Diablo Canyon Units 1 and 2 with Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events." Table 1 of this enclosure lists the action items that PGandE has already completed. Table 2 provides the schedules for work that will be performed to assure conformance with the Generic Letter. Table 3 summarizes the schedule information provided in Tables 1 and 2.

PGandE has scheduled near term completion dates for all items given high priority by the NRC. Of those, Action Items 4.1 and 4.2.1 are complete and Action Items 1.1, 4.2.2, and 4.3 are scheduled to be complete before full power operation. The Westinghouse Owners Group (WOG) has been used as a consultant to perform the generic design and licensing work for Action Item 4.3.

Performance of Action Items 2.1 and 3.1 depends on completion of the Nuclear Plant Reliability Data System (NPRDS) data acquisition project for Diablo Canyon. Presently, contract negotiations for the awarding of the NPRDS data acquisition project are in the final stages. The schedules for Action Items 2.1 and 3.1 account for installing the system, loading the data base, debugging the system, training the company personnel on the new computer system, and performing the required work.

For the medium priority action items, industry and vendor participation are being used to augment PGandE's staff in proposing a response. PGandE is participating in the Nuclear Utility Task Action Committee (NUTAC) formed on September 1, 1983 to address Action Item 2.2.2, "Vendor Interface." The NRC has acknowledged the NUTAC formation and endorsed a practical solution to Action Item 2.2.2. The NUTAC has scheduled February 1, 1984 as a release date of the final report. The schedule PGandE proposes has two inherent assumptions:

(1) That the NUTAC completes its work on schedule.

(2) That the results of the work are acceptable to both PGandE and the NRC.

In addition, the conclusions reached on Action Item 2.2.2 may have a feedback effect on Action Items 2.1 and 2.2.1. To that indeterminable extent, the schedules will be subject to change.

PGandE is also sponsoring the WOG work to aid in the response to Action Items 4.2.3, 4.2.4, and 4.5. The completion dates projected in our schedule incorporate the latest information we have obtained from WOG. For those Action Items that require pre-implementation reviews, submittals will promptly follow the acceptance by PGandE of the WOG's final recommendations.

Two of the action items in Generic Letter 83-28 do not apply to Diablo Canyon. Action Item 4.4 is specific to B&W plants and Action Item 4.5.2 is not applicable because both units at Diablo Canyon are designed to permit periodic on-line testing of the reactor trip system.

TABLE 1

GENERIC LETTER 83-28 ACTION ITEMS THAT ARE COMPLETE

Diablo Canyon currently conforms with the following action items:

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

PGandE has verified that all initial vendor recommended reactor trip breaker modifications are incorporated for both Units 1 and 2.

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

4.2.1 Periodic Maintenance.

When the generic letter was issued, the reactor trip breakers were maintained under PGandE Maintenance Procedure E-51.3, "Maintenance of All Type 480 Volt Circuit Breakers." Since that time, a new maintenance procedure, E-51.7, "Maintenance of Westinghouse Type DB 480 Volt Circuit Breakers", has been written, approved and implemented at Diablo Canyon for Units 1 and 2. This procedure incorporates the latest Westinghouse information and will be performed on a yearly basis.

TABLE 2

SCHEDULES FOR COMPLETION OF GENERIC LETTER 83-28 ACTION ITEMS

HIGH PRIORITY ACTION ITEMS

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

When Generic Letter 83-28 was issued, PGandE Administrative Procedure NPAP A-100, "General Authorities and Responsibilities of Nuclear Plant Operators" and Emergency Procedure OP-5, "Reactor Trip" were in effect. The procedures required that the Plant Superintendent determine the cause of the trip and approve plant restart. Since then, PGandE has expanded NPAP A-100 with a supplement (S1) to include the information requested in items 1-6 of Action Item 1.1. A draft of that supplement is currently being finalized. The expanded procedure is scheduled to be in effect on 12/31/83 for both Units 1 and 2. PGandE will submit a report documenting this work by 2/1/84 for both units.

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

PGandE is depending on the completion of its NPRDS as a prerequisite for performing this work. The following work is scheduled:

1. Develop a list of components required to trip the reactor. Review schematic drawings, identify and list components.

2. Develop a list of documents that identify and control activities on reactor trip system components.

3. Ensure that these documents appropriately identify these components as safety-related.

4. Develop a list of suppliers of reactor trip components.

5. Establish a formal interface with suppliers in accordance with the generic letter.

6. Establish a formal program to ensure that vendor information is current, complete, controlled, referenced, and incorporated in accordance with the generic letter.

PGandE projects that this work will be completed on 6/1/85 for Unit 1 and 6/1/86 for Unit 2. PGandE will submit a report to the NRC by 8/1/85 for Unit 1 and 8/1/86 for Unit 2 detailing the required information.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

The work required for this section is also tied to the NPRDS data acquisition project. The following work is scheduled:

1. Obtain list of reactor trip components.
2. Review maintenance procedures and Technical Specifications affecting reactor trip components to assure post-maintenance testing is performed.
3. Review vendor and Engineering recommendations to ensure appropriate test guidance is included in maintenance procedures and Technical Specifications affecting reactor trip components.
4. Identify any post-maintenance test requirements in Technical Specifications that degrade safety of reactor trip components.

PGandE projects that this work will be completed on 11/1/84 for Unit 1 and 11/1/85 for Unit 2. A statement will be submitted by 1/1/85 for Unit 1 and 1/1/86 for Unit 2 confirming implementation of items 1, 2 and 3 above. Should a Technical Specification change be required for item 4 above, a pre-implementation report will be submitted.

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

PGandE is scheduling the following work to be incorporated into the procedures discussed in our response to Action Item 4.2.1:

Develop program for trending parameters affecting operation to forecast degradation of operability.

PGandE projects that this work will be complete on 12/1/83 for Units 1 and 2.

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

PGandE has interfaced with the Westinghouse Owners Group (WOG), obtained the WOG design criteria, developed the design for modification and is currently in the process of writing the submittal to the NRC for pre-implementation review. That submittal, referencing the WOG approved design and the NRC SER issued on the generic shunt trip package, is scheduled to be submitted on 12/1/83 for Units 1 and 2. Assuming timely review and interaction, implementation of the designed modification for Units 1 and 2 is scheduled for 4/15/84.

MEDIUM PRIORITY ACTION ITEMS

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

PGandE has the following work scheduled:

1. Determine parameters required for post-trip review (coordinate with Action Item 1.1)

2. Determine what equipment will be used to obtain parameter information.

3. Determine specifications on equipment in accordance with the generic letter.

4. Determine if changes or additions to equipment are required.

This work is projected to be complete on 3/1/84 for Units 1 and 2. PGandE has scheduled the submittal of a report on this item to the NRC by 5/1/84 for Units 1 and 2.

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

2.2.1 Equipment Classification

PGandE will describe its program for ensuring that all components of safety-related systems are identified. PGandE has scheduled the following work:

1. Identify the criteria for classifying components as safety-related within systems classified as safety-related by using Engineering Procedure 3.1, "Classification of Structures, Systems and Components," Revision 3, dated 6/30/82.

2. Describe the Plant Information Management Systems (PIMS) and the methods used for its development and validation, including the NPRDS portion (NPRDS is a subset of PIMS).

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

4. Describe the management controls utilized to verify that the procedures for preparation, validation, and routine utilization of PIMS have been followed.

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

This work is scheduled for completion on 9/1/84 for Unit 1 and 9/1/85 for Unit 2.

2.2.2 VENDOR INTERFACE

PGandE is participating in the NUTAC to respond to this issue. Completion dates are projected to be 6/1/85 for Unit 1 and 6/1/86 for Unit 2. These dates are subject to the comments made in the text of this submittal. PGandE will submit a report to the NRC covering all of Section 2.2 by 8/1/85 for Unit 1 and 8/1/86 for Unit 2.

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

PGandE has the following work scheduled:

1. Review maintenance procedures and Technical Specifications affecting safety-related components to assure that post-maintenance testing is performed in accordance with the generic letter.

2. Review vendor and Engineering recommendations to ensure appropriate test guidance is included in test and maintenance procedures and Technical Specifications affecting safety-related components.

3. Identify any post-maintenance test requirements in Technical Specifications that degrade safety of safety-related components.

The completion date of this work is estimated as 6/1/85 for Unit 1 and 6/1/86 for Unit 2. PGandE will submit a statement confirming implementation of items 1 and 2 above by 8/1/85 for Unit 1 and 8/1/86 for Unit 2. Should item 3 above require a Technical Specification change, a submittal will be made before implementation.

4.2.3 and 4.2.4 REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)

In conjunction with WOG work sponsored by PGandE, the following is scheduled:

1. Develop program for life testing of breakers on acceptable sample size.

2. Develop program for periodic replacement of breakers or breaker components consistent with demonstrated life cycles.

The estimated completion dates for those activities are 2/1/84 for item 4.2.3 for Units 1 and 2 and 10/1/84 for item 4.2.4 for Units 1 and 2. PGandE will submit a pre-implementation report for both items and units by 12/1/84.

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

4.5.1 PGandE plans to perform online functional testing of undervoltage and shunt trip features by 5/1/84 for Units 1 and 2. This date is consistent with plans to have Action Item 4.3 implemented by 4/15/84. PGandE will submit a statement to the NRC confirming the completion of this action by 7/1/84 for both units.

4.5.3 PGandE is sponsoring WOG work to perform the following:

Review existing intervals for online testing required by Technical Specifications to determine that intervals are consistent with high system availability, as required by the generic letter.

The estimated completion date is 12/1/84 for Units 1 and Unit 2. At that time, a pre-implementation report will be submitted should a Technical Specification change be required.

TABLE 3

SUMMARY OF SCHEDULES FOR GENERIC LETTER 83-28 ACTION ITEMS

(DERIVED FROM TABLES 1 AND 2)

Action Item	Scheduled Completion Date Unit 1	Unit 2
High Priority Items		
1.1	12/31/83	12/31/83
2.1	05/01/85	06/01/86
3.1	11/01/84	11/01/85
4.1	Complete	Complete
4.2.1	Complete	Complete
4.2.2	12/01/83	12/01/83
4.3	04/15/84	04/15/84
Medium Priority Items		
1.2	03/01/84	03/01/84
2.2.1	09/01/84	09/01/85
2.2.2	06/01/85	06/01/86
3.2	06/01/85	06/01/86
4.2.3	07/01/84	07/01/84
4.2.4	10/01/84	10/01/84
4.4	N.A. (1)	N.A. (1)
4.5.1	05/01/84	05/01/84
4.5.2	N.A. (2)	N.A. (2)
4.5.3	12/01/84	12/01/84

1. Applicable only to B&W plants.

2. Not Applicable, because DCPP design permits on-line testing.