LILCO LONG ISLAND LIGHTING

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

March 9, 1984 SNRC-1013

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Response to Generic Letter 83-28
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference 1: LILCO letter, SNRC 960 (J. L. Smith) to the NRC (H. R. Denton), dated September 9, 1983.

Dear Mr. Denton:

The attached report is in response to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events". This report addresses each NRC position and provides a description of the actions taken by LILCO in response. Where a program is under development an estimated commitment date has been provided.

If any additional information is required, please contact this office.

Very truly yours,

B R McCaffrey

B. R. McCaffrey
Manager, Nuclear Compliance and Safety

RJT:ck

cc: C. Petrone
All Parties Listed in Attachment I

Attachment

8403200302 840309
PDR ADOCK 05000322
A PDR

A055
1/40

FC-8935

ATTACHMENT I

Bernard M. Bordenick, Esq.
David A. Repka, Esq.
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Mr. James Dougherty
3045 Porter Street
Washington, D.C. 20008

Ben Wiles, Esq.
Assistant Counsel to the Governor
Executive Chamber
State Capitol
Albany, NY 12224

Mr. Marc W. Goldsmith
Energy Research Group
4001 Totten Pond Road
Waltham, Massachusetts 02154

Martin Bradley Ashare
Suffolk County Attorney
H. Lee Dennison Building
Veterans Memorial Highway
Hauppauge, NY 11788

Ralph Shapiro, Esq.
Cammer and Shapiro, P.C.
9 East 40th Street
New York, NY 10016

Herbert H. Brown, Esq.
Lawrence Coe Lanpher, Esq.
Karla J. Letsche, Esq.
Kirkpatrick, Lockhart, Hill
Christopher & Phillips
8th Floor
1900 M Street, N.W.

Matthew J. Kelly, Esq.
State of New York
Department of Public Service
Three Empire State Plaza
Albany, NY 12223

Gerald C. Crotty, Esq.
Counsel to the Governor
Executive Chamber
State Capitol
Albany, NY 12224

MHB Technical Associates
1723 Hamilton Avenue
Suite K
San Jose, CA 95125

Stephen B. Latham, Esq.
Twomey, Latham & Shea
33 West Second Street
P.O. Box 398
Riverhead, NY 11901

SNRC

LILCO Response to Generic Letter 83-28

Item 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.

LILCO Response

LILCO has developed a systematic post-trip review program that will be implemented prior to the operation of the Shoreham Nuclear Power Station (SNPS). The program ensures that any unscheduled reactor shutdown is analyzed to determine if the plant can be restarted safely. The controlling procedure for this program is Station Procedure 21.003.01, "Operating Reports." This procedure is attached as Appendix A.

The following is an itemized description of the SNPS post-trip review program in response to NRC Generic Letter 83-28.

Item 1.1.1 NRC Request - Describe the criteria for determining the acceptability of restart.

LILCO Response

LILCO is participating in the BWR Owner's Group (BWROG) Salem ATWS, Generic Issues Committee effort to develop a generic description of the criteria for determining the acceptability of plant restart after an unscheduled reactor shutdown. Although LILCO expects to endorse the committee's position, we will submit a formal response to the NRC within sixty (60) days after the receipt of the finalized BWROG response to allow time for a Shoreham-specific evaluation.

Item 1.1.2 NRC Request - Describe the responsibilities and authorities of personnel who will perform the review and analysis of these events (unscheduled reactor shutdowns).

LILCO Response

The Watch Engineer is responsible to ensure that the plant is operated safely and in accordance with the requirements of the facility Operating License, Technical Specifications, and approved operating procedures. The Watch Engineer has the responsibility and authority to direct a shutdown of the reactor whenever he determines that the safe operation of the plant is in immediate jeopardy or when operating parameters exceed reactor protection set points and automatic action does not occur. The Watch Engineer reports to the Operating Engineer and is responsible to ensure data is collected and an analysis of such data performed to determine the cause of any unscheduled shutdown. He analyses items such as equipment malfunctions, procedure inadequacies and operating errors to determine the cause of the scram. He does not recommend a reactor restart unless the cause of the scram is fully understood.

The Shift Technical Advisor is responsible for advising the Watch Engineer regarding reactor core damage prevention or mitigation, during plant accident or transient conditions. He is responsible to ensure the required data is collected and an analysis is done of the data to determine the cause of the event.

The Operating Engineer is responsible for directing day to day operation of the Shoreham Nuclear Power Station unit including startup, operation, and shutdown of all equipment in accordance with approved operating procedures, Technical Specifications and regulatory requirements. In addition, the Operating Engineer has the authority to order the shutdown of the reactor whenever he determines that the safe operation of the plant might be jeopardized or if it appears that operating parameters will exceed the reactor protection setpoints. The Operating Engineer ensures that the unscheduled shutdown has been analyzed and the cause determined and corrected prior to authorizing a restart of the reactor.

Item 1.1.3 NRC Request - Describe the necessary qualifications and training for the responsible personnel.

LILCO Response

LILCO is participating in the BWR Owner's Group (BWROG) Salem ATWS Generic Issues Committee effort to develop a generic description of the necessary qualifications and training for the responsible personnel. Although LILCO expects to endorse the committee's position, we will submit a formal response to the NRC within sixty (60) days after the receipt of the finalized BWROG response to allow time for a Shoreham-specific evaluation.

Item 1.1.4 NRC Request - Describe 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 Item 1.2).

LILCO Response

The primary sources of information used to evaluate unscheduled reactor shutdowns are the plant process computer and various chart recorders used to record specific plant parameters.

The process computer has the capability of measuring and sorting values of analog variables at various time intervals to provide a post trip log of historical data. This data is automatically printed following a reactor scram. The plant process computer and additional sources of plant information are discussed in the response to Item 1.2, "Post Trip Review - Data and Information Capability".

Item 1.1.5 NRC Request - Describe 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).

LILCO Response

LILCO is participating in the BWROG Salem ATWS Generic Issues Committee effort to develop generic methods and criteria to aid in the comparison of the event generated information with known or expected plant behavior. Although LILCO expects to endorse the committee's position, we will submit a formal response to the NRC within sixty (60) days after the receipt of the finalized BWROG response to allow time for a Shoreham-specific evaluation.

Item 1.1.6 NRC Request - Describe 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 re-start) and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

LILCO Response

The Operating Engineer will implement SP 21.003.01 and request that the Review of Operations Committee (ROC) be convened when the cause of the event cannot be positively identified. The ROC will use the same information as the Watch Engineer, Shift Technical Advisor and Operations Engineer to make their determination. In addition, members of the Shoreham Plant Staff can enlist the expertise of other personnel in the Office of Nuclear or an outside organization to help in determining the cause of the event.

Item 1.1.7 NRC Request - Items 1.1.1 through 1.1.6 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.

LILCO Response

Station Procedure 21.003.01, "Operations Reports", contains the Scram Report and Scram Evaluation procedures. These procedures provide a systematic method to assess any unscheduled reactor shutdown.

Item 1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

NRC Position - 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.

LILCO Response

At SNPS, plant data and information that may be used to diagnose the cause of an unscheduled reactor shutdown is contained in the plant process computer, the Emergency Response Facility (ERF) computers, radiation monitoring computers and on strip chart recorders. Due to the long engineering lead time required for the ERF computer system, the function will be incorporated in two phases. Phase I will utilize an upgraded plant process computer to provide a limited Safety Parameter Display System SPDS and a special data logging function. Phase I will be available at fuel load. Phase II will be the permanent system. Phase II will be fully operational within six (6) months after the first refueling outage.

Item 1.2.1 Capability for Assessing Sequence of Events (On-Off Indications)

Item 1.2.1.1 NRC Request - Provide a brief description of the equipment.

LILCO Response

The capability for assessing sequence of events (on-off indications) at Shoreham is provided mainly by the balance-of-plant (BOP) and nuclear steam supply system (NSSS) sequence of events logs in the plant process computer and supplemented by the digital parameters which provide inputs to the ERF Phase II computer system. The BOP sequence of events log monitors those digital points which lead directly to or directly cause a turbine trip. The NSSS sequence of events log performs a similar function for digital points which directly lead to or cause a reactor scram. The ERF Phase II computer system is briefly summarized in Section 1.2.3.

Item 1.2.1.2 NRC Request - Describe the monitored parameters.

LILCO Response

Appendix B provides a cross-reference relating plant process computer sequence of events computer points with associated plant annunciator alarm points. The distinction between NSSS and BOP points can be made by noting the system number column. System 1C51 and 1C71 are NSSS points and the remainder are BOP points.

Item 1.2.1.3 NRC Request - Describe the time discrimination between events.

LILCO Response

The plant process computer sequence of events log provides a two-millisecond resolution between events.

Item 1.2.1.4 NRC Request - Describe the format for displaying data and information.

LILCO Response

The plant process computer sequence of events logs are displayed on a printer in the main control room. Appendix C is a sample of the format of the sequence of events log.

Item 1.2.1.5 NRC Request - Discuss the capability for retention of data and information.

LILCO Response

Since the plant process computer system provides hard copy outputs, the data and information can be retained indefinitely.

Item 1.2.1.6 NRC Request - Describe the power source(s).

LILCO Response

The plant process computer is normally powered from non-Class 1E inverters (uninterruptible power) which is backed by diesel generator power.

Item 1.2.2 Capability for Assessing the Time History of Analog Variables.

Item 1.2.2.1 NRC Request - Provide a brief description of the equipment (e.g., plant computer, dedicated computer, strip charts).

LILCO Response

The capability for assessing the time history of the analog variables which may be used to determine the cause of an unscheduled reactor shutdown and the functioning of safety-related equipment is provided by the analog parameters monitored by the process computer system, the ERF Phase I portion of the process computer system and by analog parameters recorded and displayed on various strip chart recorders located in the main control room. Additionally, radiological and meteorological parameters are available via the computer-based radiological monitoring system. Analog signals from Category I (safety related) radiation monitors are also recorded on strip chart recorders in the main control room.

Item 1.2.2.2 NRC Request - Describe the parameters monitored, sampling rate, and basis for selecting parameters and sampling rate.

LILCO Response

An analog parameter post-trip log is generated automatically by the plant process computer upon detection of a unit trip. The log consists of ten (10) NSSS parameters and forty (40) BOP parameters which are accumulated for a period of five (5) minutes before and after a unit trip. Appendix D lists the NSSS and BOP parameters printed in the process computer post-trip log. To provide a limited SPDS/ERF capability during initial plant operation (Phase I), the process plant computer has been upgraded to include some SPDS graphic display and Technical Support Center (TSC) post-trip logging capability. Appendix E contains a list of points which are available for this purpose. Analog signals sent to strip chart recorders are recorded continuously. The following analog parameters are recorded in the control room:

(a) Reactor level
(b) Reactor pressure
(c) Suppression pool temperature
(d) Signals from QA Category I radiation monitors
(e) Reactor building differential pressure
(f) Drywell pressure
(g) Suppression pool level
(h) Neutron flux

Radiological and meteorological parameters are provided by the radiological monitoring system. During ERF Phase I, the data will be resident only in the radiological monitoring system computers and will be available upon demand. Appendix F provides a list of the monitored points.

Item 1.2.2.3 NRC Request - Describe the duration of the time history (minutes before trip and minutes after trip).

LILCO Response

The strip chart recorders listed above record continuously before and after the trip. The analog post-trip log function of the plant process computer provides output for the NSSS parameters at five (5) second intervals and output for the BOP parameters at fifteen (15) second intervals. Data is provided for five (5) minutes before and after a unit trip in a tabular format. The ERF Phase I Computer provides three types of TSC logs upon demand. They are:

1. Two hours pre-event data of 112 points at one minute intervals plus five minutes post-event data which includes:

    a. 9 points every second

    b. 16 points every 15 seconds

    c. 87 points every minute

2. 5 minutes pre-event to 5 minutes post-event data, consisting of:

    a. 9 points every second

    b. 16 points every 15 seconds

    c. 87 points every minute

3. Continuous TSC log of 112 points per minute.

The radiological monitoring computer can provide historical data upon demand as follows:

1. Sixty (one second intervals) listings of instantaneous values of all points in one minute data blocks.

2. Sixty one-hour averages for each point based upon data gathered in 1 above.

3. Sixty forty-eight hour averages for each point based upon data gathered in 2 above.

1.2.2.4 NRC Request - Describe the format for displaying data including scale (readability) of time histories.

LILCO Response

Analog parameters monitored on one, two, or three pen strip chart recorders having 4 inch chart paper (minimum) are displayed on paper which is scaled for the parameter being recorded. Additionally, a separate scale and pointer on the recorder shows the current value of the parameter being recorded.

Analog parameters monitored by the ERF Phase I computer system can be displayed on CRT consoles as shown in Appendix G. The format for the ERF Phase I Special Log is shown in Appendix H.

1.2.2.5 NRC Request - Describe the capability for retention of data, information and physical evidence (both hardware and software).

LILCO Response

Strip chart recorder output is available indefinitely as long as the chart rolls are stored in an accessible location. This is true also for printed outputs from the plant process computer system (including ERF Phase I). The ERF Phase I computer system utilizes periodic data transfers to magnetic tape drives for long-term data storage. Therefore, the data is expected to be available indefinitely.

1.2.2.6 NRC Request - Describe the power source(s).

LILCO Response

The plant process computer system, including ERF Phase I, is powered by non-Class 1E inverters (uninterruptible power) and is backed by diesel generator power. The radiation monitoring system computers are also powered from diesel backed, non-Class 1E power. Recorders for parameters a, b, c, d, e, f, and g listed in Section 1.2.2.2 above are powered from 1E power. Recorders for parameter h are powered from non-Class 1E sources. In the case of parameter c, loop power for the 1 foot temperature sensors is from RPS power, while the loop power for the 2 foot temperature sensors is from Class 1E (Divisions I and II) power sources.

Item 1.2.3 NRC Request - Describe other data and information provided to assess the cause of unscheduled reactor shutdown.

LILCO Response

The ERF Phase II computer system will replace the Phase I system within six (6) months after the return from the first refueling outage. The Phase II system is essentially independent of the Phase I system and the SPDS and special data logging functions of Phase 1 will not be affected during the startup of the Phase II system. The information provided is preliminary and is being used for development of the ERF Phase II System per SNRC-863 (4/4/83).

The plant process computer will remain the major source of detailed sequence of events information since most of these points will not be monitored by the Phase II system. Contact (digital) inputs to the ERF Phase II computer system are scanned with a one-second resolution.

The ERF Phase II computer system digital point status may be displayed on cathode ray tube (CRT) terminals and printers in the technical support center (TSC) and emergency operations facility (EOF) and by CRT only in the control room. The ERF post-trip log concept under development will consist of the entire data base taken at various frequencies and intervals. Printed outputs can be retained indefinitely. The ERF Phase II system will be powered from Non-Class 1E inverters (uninterruptible power).

The ERF Phase II system will implement full SPDS and analog trend information as described below. In addition, a data link will transfer certain data from the radiation monitoring system to the ERF computer to allow display of certain meteorological and radiological parameters.

These points in the input list were selected using NUREG-0696 guidelines with Shoreham specific Regulatory Guide 1.97 BWR parameters comprising the minimum data set. The scan class was selected based on the rate at which an individual parameter could change so as not to lose information during transient conditions. The scan periods vary from 0.1 to 60 seconds.

The ERF Phase II computer system records continuously on a circular file and stores data for two hours prior to any reactor scram (manual or automatic) and for up to two weeks following reactor scram. This feature is effective for both analog and digital inputs. The capability to record up to two weeks of additional post-event data is provided by utilizing disk drives for long-term data storage. Data recording beyond two weeks is possible by transferring the data on disk to magnetic tape.

Further detailed information on the ERF Phase II Computer System is outlined in the Shoreham FSAR, Section III.A.1.2. It should be noted that there are additional digital and analog parameters which are monitored by the plant process computer and additional analog recorders in the control room which do not specifically provide sequence of events information but which do give information which could be useful in assisting in the determination of the cause of a reactor trip.

LILCO may modify the number of points monitored, listed setpoints, scan groups, formats, etc., based upon the results of startup and operations experience, to improve the usefulness of the monitored data.

Item 1.2.4 NRC Request - Provide the schedule for any planned changes to existing data and information capability.

LILCO Response

The plant process computer and ERF Phase I will be operational at fuel load. ERF Phase II will be fully operational within six (6) months after the first refueling outage. It should be noted that the Phase I and II systems are completely independent but use some equipment in parallel.

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

Item 2.1.1 Equipment Classification (Reactor Trip System Components).

NRC Position - Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement.

LILCO Response

LILCO has identified the Reactor Trip System (RTS) components that should be classified as safety related for SNPS. The resulting list of active components of existing plant systems that function to implement a reactor scram is in the process of being incorporated into the SNPS Composite Component List (CCL). The CCL is a listing of safety-related components that is part of the plant information handling system. It is further discussed in the response to item 2.2.1.

In addition, LILCO is participating in the BWROG Salem ATWS Generic Issues Committee effort to develop a generic Reactor Trip Function List and provide several safety classification examples for components with various complexity levels. LILCO expects to endorse the Committee's position and we will submit a formal response to the NRC within sixty (60) days after the receipt of the finalized BWROG response. This response will indicate the status of any CCL modifications deemed necessary as a result of the Shoreham-specific evaluation of the BWROG generic response.

Item 2.1.1 Vendor Interface (Reactor Trip System Components).

NRC Position - 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 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. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of

positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and non-nuclear 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.

LILCO Response

LILCO is instituting a program to receive, control, review and utilize vendor technical information such as GE Service Information Letters (SILS) and Service Advisory Letters (SALS) as per the requirements of Corporate Procedure NED 2.07, "Review of Vendor Technical Bulletins". The procedure is expected to be implemented by April, 1984. GE also has further responsibilities, pursuant to 10 CFR 21, which require it to identify safety problems.

LILCO also reviews relevant industry information and experience through programs involving participation in the Nuclear Plant Reliability Data System (NPRDS) and the Significant Event Evaluation and Information Network (SEE-IN), both of which are managed in INPO.

LILCO is participating in the BWROG Salem ATWS Generic Issues Committee effort to develop a program to assure that RTS Vendor information is complete and current. Although LILCO expects to endorse the Committee's position, we will submit a formal response to the NRC approximately sixty (60) days after the receipt of the finalized BWROG response to allow time for a Shoreham-specific evaluation.

In addition, LILCO is a member of the Nuclear Utility Task Action Committee (NUTAC) which has been formed on vendor interface. An approved program is expected at the end of the first quarter of 1984. The NUTAC program is further discussed in the response to item 2.2.2.

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

Item 2.2.1 Equipment Classification (Programs For All Safety-Related Components)

NRC POSITION - 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.

Item 2.2.1.1 NRC Request - Describe 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.

Item 2.2.1.2 NRC Request - Provide 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.

LILCO Response (Items 2.2.1.1 and 2.2.1.2)

Components within structures or systems classified as safety related are designated as "safety related" if they are necessary to assure:

1) The integrity of the reactor coolant pressure boundary,

2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or

3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100, Appendix A.

Accordingly, two broad functional classifications; safety-related and non safety-related, have been established. QA Category I is assigned to safety-related components. QA Category II is assigned to nonsafety-related components.

Inherent in the designation of a component's classification is a recognition of the function performed by the component. The General Design Criteria, Appendix A to 10 CFR Part 50, establishes the minimum requirements for the design of a nuclear power plant. The American Nuclear Society Standard ANS 22, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants", was used in establishing the classification of structures, systems and components for SNPS. An evaluation of the design bases of the Shoreham Nuclear Power Station as measured against the General Design Criteria, is presented in Section 3.1 of the Shoreham FSAR. A summary of the classification of the Shoreham structures, systems and components is presented in Section 3.2 of the Shoreham FSAR.

LILCO has a comprehensive program which ensures that all safety-related components are identified as safety-related on documents, procedures and information handling used in the plant to control safety-related activities. The program reflects the design criteria presented in the Shoreham FSAR, the LILCO Quality Assurance Program in effect during construction and operation, and the LILCO procedures controlling all engineering and operational activities at Shoreham. LILCO has performed a Plant Configuration Review to ensure that the as-built configuration of safety-related systems conform to the commitments in the FSAR and licensing documents. The implementation and the results of the SNPS Plant Configuration Review Program were recently reviewed and found acceptable by the NRC Office of Inspection and Enforcement, Region I, as documented in Inspection Report 50-322/83-38. In addition, LILCO has a program in place to ensure that by the time of fuel load, or shortly thereafter, the configuration of the plant will be accurately reflected by drawings that will be used by the station operations staff.

Drawings which address safety-related components are identified as such and labeled QA Cat 1.

In addition to the engineering assurance and design control efforts implemented for Shoreham, LILCO has maintained a quality assurance program which meets the requirements of 10 CFR Part 50, Appendix B. The quality assurance program will be applied to structures, systems and components throughout operations, maintenance, station modifications and appropriate repair activities at Shoreham. The operational QA program will apply to all organizations performing design, design review and/or design audit activities. Section 3 of the LILCO QA manual describes the QA program requirements established to provide this control. Similarly, Section 4 of the LILCO QA manual describes the QA program requirements established to control procurement of safety-related material, equipment and services, while Section 5 of the LILCO QA manual assures that activities affecting the quality of safety-related structures, system and components during operations are controlled according to instructions, procedures and drawings.

Finally, LILCO has developed a comprehensive set of station procedures to ensure that the design controls and criteria are maintained throughout the operational life of Shoreham. These procedures address the procurement of spare or replacement parts, the installation, inspection and testing of all components and the performance of all maintenance activities.

The identification system, including unique part or mark numbers, developed during the design and construction phases, is maintained current during the operational phase. Identification is provided on specifications, drawings, purchase orders or other documents, maintaining traceability to manufacturing and inspection documents, heat numbers, and all test reports. The component identification is either on the item or on record, directly and readily traceable to the item. Physical identification is used to the maximum extent possible, and applied so it does not affect the item's function. The identification is verified throughout fabrication, assembly, shipping, and installation. The operating plant staff will maintain the identification system used during the design and construction phase.

To facilitate operations, LILCO developed a Composite Component List (CCL). The CCL will supplement existing engineering and operations controls. The CCL contains the unique component mark number, the applicable procurement or construction specification reference and the records management file references. This list was developed based on the engineering, construction and preoperational test activities performed for Shoreham. In order to ensure the accuracy of the list, LILCO has performed an engineering review of the list and developed procedures to control inputs and changes to the list.

Item 2.2.1.3 NRC Request - Provide 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.

LILCO Response

LILCO uses approved procedures for ensuring that safety-related systems, structures, and components are identified, and are treated as such for various plant activities such as design modifications, maintenance, surveillance, parts replacement, repair handling, inspection, etc. When for any activity, a uniquely identifiable component safety classification must be verified, the Nuclear Engineering Department controlled Composite Component List is consulted to determine if the component is safety or nonsafety-related.

Appropriate guidance is given in SP 12.013.01, "Maintenance Work Requests" for maintenance activities performed on safety-related systems, structures, and components. The Scheduled Activity Worksheets, (SAWS), used for implementation of surveillance and preventive maintenance programs, also indicate whether the activity is to be performed on safety or nonsafety-related components for utilization of applicable Quality Assurance controls.

Safety related replacement components and parts have initially been evaluated to be compatible with the originally supplied components and parts per the requirements of the applicable Safety Related Purchase Specifications. New components (added as a result of plant modifications will receive an independent engineering and Quality Assurance evaluation for determining their safety classification, safety function, and failure effect on other safety related components as described in the LILCO Program Description PD-NE-01, "Nuclear Organization Interim Management Control Program for Station Modifications". If a component is determined to perform a safety function, it will also receive a detailed review of the technical, packaging, shipping, tagging, traceability/documentation, vendor qualification, etc. requirements. The above engineering and Quality Assurance evaluation and technical review shall also be performed before an existing plant component can be upgraded or downgraded from its original safety classification.

Approved procedures are used for receiving inspection, storage, and issue of safety related spare parts, materials, and components.

Item 2.2.1.4 NRC Request - Describe the management controls utilized to verify that the procedures for the preparation, validation, and routine utilization of the information handling system have been followed.

LILCO Response

The LILCO QA Program requires that activities affecting safety are accomplished and controlled in accordance with documented instructions and procedures. To comply with this requirement the LILCO Nuclear Operations Support and Nuclear Engineering Departments have prepared procedures. These procedures describe the preparation, revision and control of the component classification designation system and provide

a description of the criteria used to classify structures, systems and components. The procedures also require classifications to be applied to all structures, systems and components that may be added due to modifications, repairs and additions to ensure that the quality of the plant is not degraded during its operating life.

The classification of the items mentioned above are listed in the Composite Component List (CCL). Procedures have been prepared to describe the method used to revise the CCL. These procedures require any changes to be approved and documented.

In addition, the QA Audit Program described in Section 18 and Appendix A of the QA Manual identifies auditable functions of the Nuclear Operations Support and the Nuclear Engineering Department who prepare, review and control the CCL. All audits are performed in accordance with QA Procedures.

Item 2.2.1.5 NRC Request - Demonstrate 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.

LILCO Response

The demonstration that adequate design verification and qualification testing has been in place for safety related components for Shoreham was achieved through rigorous equipment qualification programs. These programs include 1) Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment, and 2) Environmental Qualification of Safety-Related Electrical Equipment. Both of these SNPS programs have received extensive review by the NRR Staff and their consultants. The extent of these reviews, which consisted of programatic reviews and also audits of staff selected program documentation, is identified in the Shoreham Safety Evaluation Report (SER), NUREG-0420 and its supplements. These specific programs are addressed in Sections 3.10 and 3.11 of the SER. These programs demonstrated the adequate procurement, testing and documentation of these requirements. Required service conditions were identified for each component along with verification that the qualification documentation adequately handled the service conditions required. Documentation packages have been assembled for all components requiring seismic and/or environmental qualification the documentation package contains the following:

Environmental Qualification

1) Index, which fully identifies the content of the package by name, document number, forms, dates, revision number, titles and/or summaries.

2) Required Service Condition form(s) applicable to the qualification package.

3) Environmental Qualification Summary Sheet summarizing the qualification documentation review.

4) Environmental Qualification Report Evaluation form; a checklist which guides the document review assessment of the qualification documentation.

5) Stone & Webster Engineering Corporation Review of Supplier's Technical Document form 5040.51B, where applicable.

6) Test Report

7) Applicable supplemental analysis.

8) Pertinent backup correspondence (eg. vendor letters, S&W letters, LILCO letters), and supplemental documentation.

Dynamic Qualification

1) Index of included and referenced documents.

2) Reviewers summary of qualification basis.

3) Completed NRC form entitled "Qualification Summary of Equipment," including required acceleration level or response spectra, as appropriate.

4) Qualification report(s).

The maintenance of qualified life of these safety-related components is ensured through the Shoreham surveillance and maintenance programs. The Shoreham surveillance and maintenance programs include documented program plans and procedures, to assure that the safety-related equipment is maintained in a state of readiness and operability so that it will perform its intended safety function upon demand.

The results of the Environmental Qualification Program are directly input to the Shoreham maintenance and spare parts programs to ensure timely device or parts replacement as needed due to identified qualified life limitations and to ensure timely performance of any other identified maintenance activities required to preserve the applicability of the qualification.

The Shoreham surveillance and maintenance program includes information supplied by the equipment manufacturers and vendors regarding required equipment maintenance actions and their frequency.

The equipment environmental qualification packages are reviewed by the Station Technical Support Staff and sub-packages of relevant information assembled and distributed to station Maintenance and Instrument & Controls sections as appropriate. Upon receipt of these packages, the responsible groups arrange for any new information to become part of the computerized program for plant maintenance which will alert them to upcoming maintenance requirements on a timely basis.

Reporting of equipment failures will be performed by the LILCO Nuclear Engineering Department (NED) using the Nuclear Plant Reliability Data System (NPRDS) program administered by the Institute of Nuclear Power Operations (INPO). During 1981, plans were completed within LILCO to initiate the data collection and reporting effort. These plans have been completed and are now being updated in a manner consistent with the Plant Modification Program. Procedures will be prepared and approved outlining the responsibilities for the accurate and uninterrupted flow of data from the Shoreham Nuclear Power Station to the NPRDS.

The Shoreham Equipment Qualification Programs are on-going and continue to be in place as equipment is modified or additional equipment added to the plant design. Documentation packages are either revised to reflect the changes or new documentation packages are created. These packages receive extensive review to ensure addition of any applicable new requirements to the Station Surveillance and Maintenance Programs.

Procurement

LILCO has currently in place a Corporate Policy to control procurement of material, equipment or services for Shoreham. This Policy, NOC 4 (Nuclear Operations Corporate Policy) entitled, "Corporate Procurement Document Control" identifies programatic requirements which include the requirement that all procurement activity. The technical review ensures that appropriate technical requirements are specified including seismic or environmental qualifications.

Each LILCO department with procurement responsibility has specific departmental implementing procedures in place to ensure compliance with the Corporate Policy. For procurement activities performed by S&W as LILCO's agent, S&W Project Procedures are in place to ensure appropriate technical requirements are specified. The S&W procurement documents are further reviewed by LILCO in accordance with other LILCO procedures to ensure the documents are technically adequate.

Maintenance of Design Adequacy and Qualification

To ensure adequate control of future design modifications at Shoreham, a Design Control Program was developed and is in place. This Program, entitled "LILCO Nuclear Organization Interim Management Control Program for Station Modifications" controls all modifications to the Shoreham plant from initiation through implementation and final closeout. It requires that all associated documents and procedures are updated to reflect specific plant modifications. The NRC Office of Inspection and Enforcement, Region I, has reviewed this program in detail and reported favorably in Inspection Report 50-322/83-29.

Component Classification

As new components are added or existing components modified (via the Design Modification Program) their classification is identified (i.e, safety related, nonsafety-related) and they are added to the Composite Component List (CCL). The CCL is a computerized index of uniquely identified components for the Shoreham facility. Additions, deletions or changes to this list are procedurally controlled by the LILCO Nuclear Engineering Department.

This list ensures that procurement and maintenance activities required for safety related components are followed for items designated as safety related. The on-going Equipment Qualification Programs are maintained to identify additions, deletions or changes and documentation packages are either revised or prepared to

reflect these new requirements. This ensures that all required documentation was received, reviewed and is adequate to represent proper qualification.

Item 2.2.1.6 NRC Request - 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").

LILCO Response

The Shoreham Nuclear Power Station program for the classification of safety related components is described in Sections 2.2.1.1 through 2.2.1.5 of this submittal. With regard to the equipment classification program in use at Shoreham for structures, systems and components important to safety, we are participating in the Utility Safety Classification Group and are seeking a generic resolution to the staff's concern in this regard through the efforts of this Group. We do not agree that plant structures and components important to safety constitute a broader class than the safety related set. As previously stated in SNRC-844 (3/2/83) and SNRC-854 (3/8/83) LILCO believes 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.

Item 2.2.2 NRC Position - 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 license 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.

LILCO Position

LILCO supported the BWROG recommendation to participate in a utility sponsored Nuclear Utilities Task Action Committee (NUTAC) to address the NRC concerns raised

in this item. To date LILCO has fully participated in this committee activity which has resulted in a Vendor Equipment Technical Information Program (VETIP) to be managed by INPO. Several VETIP drafts have already been issued by INPO. A utility/INPO approved VETIP is expected at the end of the first quarter of 1984. Although LILCO fully expects to endorse the INPO issued VETIP, LILCO intends to withhold formal endorsement for 60 days in order to allow time for a Shoreham-specific evaluation

Item 3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

Item 3.1.1 NRC Request - Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-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.

LILCO Response

LILCO is committed to the requirements of the Technical Specifications to operate the Shoreham Nuclear Power Plant. The Technical Specifications outline the acceptable operating parameters, Limiting Conditions for Operation (LCO), and surveillance requirements for the safety related systems, instrumentation channels and components of the plant. This includes the instrumentation channels and components of the Reactor Protection System. In addition to the regular intervals specified in the Technical Specifications, surveillance testing (functional operability test) is required to be performed on a system, instrumentation channel or component, each time the component is subject to a maintenance activity prior to returning the component or system to service. Station procedure SP 12.013.01, "Maintenance Work Requests", (MWR) outlines the requirements for post- maintenance operability tests ("Postwork Tests") following a component maintenance or repair. The required post-maintenance operability test is determined by the Operations Section and documented on the MWR form. In most cases, the demonstration of the post-maintenance operability of a component will be satisfied by the performance of a Technical specifications surveillance test (i.e. functional operability test). The test may be performed by the Maintenance, I&C Operations or other sections (depending on the department jurisdiction over the test), however, the verification of implementation is the Operations Section's responsibility.

An Operations Section procedure SP 21.001.02, "Return of Safety Related Components to an Operable Status", supplementing SP 12.013.01, "Maintenance Work Requests", was also established to improve the method by which MWRs will be handled for retests. This procedure extends the post-maintenance operability test requirements for all safety related components even if there is no specific surveillance testing specified by the Technical Specifications or the Inservice Inspection Program. The procedure also gives guidelines for determining the extent of post-maintenance operability testing for the various plant components.

Item 3.1.2 NRC Request - 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.

LILCO Response

Appropriate vendor and engineering recommendations have been or are being incorporated in the test and maintenance procedures and Technical Specifications. Department level procedures require the use of engineering documents and vendor manuals for the generation of the various test and maintenance procedures. These source materials are included in the reference section of the procedures.

The LILCO program calls for periodic review and updating of the station operating, test and maintenance procedures to assure that the latest vendor and engineering recommendations are incorporated and industry practices are followed. Station Procedure SP 12.007.01, "Technical Correspondence and Bulletins," assures that all incoming correspondence bulletins, and vendor information are properly tracked and assigned for action within the Shoreham Plant Staff and the necessary updates to existing procedures and programs are made, as appropriate.

Item 3.1.3 NRC Request - 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 function testing.)

LILCO Response

The Shoreham Technical Specifications are currently undergoing review and approval. If, during our review and use of the SNPS Technical Specifications any requirements are found to degrade rather than enhance safety, the appropriate changes and associated justification will be submitted to the NRC.

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

Item 3.2.1 NRC Request - 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.

Item 3.2.2 NRC Request - Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical specifications were required.

Item 3.2.3 NRC Request - 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.

LILCO Response (Items 3.2.1, 3.2.2, and 3.2.3)

The response to Item 3.1 is also applicable to all other safety-related components.

Item 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.

Item 4.5.1 NRC Request - The diverse trip features to be tested include the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

Item 4.5.2 NRC Request - 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.

LILCO Response (Items 4.5.1 and 4.5.2)

LILCO is participating in the BWROG Salem ATWS Committee effort to address the NRC concerns raised by these items. GE has been requested to address these concerns and has issued several draft responses which have received internal committee review. A final Committee position on these items is expected during the second quarter of 1984. Although LILCO expects to endorse the Committee's position, we will submit a formal response to the NRC within sixty (60) days after the receipt of the finalized BWROG position to allow time for a Shoreham-specific evaluation.

Item 4.5.3 NRC Request - 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.

LILCO Response

LILCO is currently evaluating the proposed BWROG Technical Specification Improvement Committee effort with regard to item 4.5.3 in parallel with an internal review of the proof and review copy of the SNPS Technical Specifications. A decision regarding participation in the BWROG Tech Spec Committee will be made within three (3) months.

APPENDIX A

Submitted: W2 Gunther
Reviewed/OQA Eng.: J. Thomas Rose
Approved: W.C. Kluge Jr.
(Plant Manager)

MC-1

SP Number	21.003.01
Revision	1
Date Eff.	12/21/83
TPC	
TPC	
TPC	

OPERATIONS REPORTS

1.0 PURPOSE

This procedure describes measures which operations personnel will use to inform appropriate levels of management of Reactor Scrams and events which may require reporting to outside agencies.

2.0 RESPONSIBILITY

The Operating Engineer shall be responsible for assuring implementation of the requirements of this procedure.

3.0 DISCUSSION

3.1 It is the responsibility of all personnel in the Operations Section to report conditions that may have an adverse effect on the plant or its operation to appropriate levels of management. There are several methods operating personnel have to report these conditions such as:

3.1.1 Maintenance Work Requests (MWR) - Reference 11.2
3.1.2 LILCO Deficiency Report (LDR) - Reference 11.10
3.1.3 Scheduled Activity Worksheets (SAWS) - References 11.3 and 11.4
3.1.4 Operating Logs - Reference 11.5
3.1.5 Scram Reports - this procedure
3.1.6 Appendix 12.1 Scram Report - this procedure
3.1.7 Appendix 12.2 Scram Evaluation - this procedure

3.2 The selection of one of the above control mechanisms for any particular condition or event will depend on the nature of the event and to some extent the preference of the individual responsible for the documentation and followup action.

3.3 This procedure will discuss the use of the Scram Report and the LILCO deficiency Report.

3.3.1 Scram Report - Appendix 12.1

A report used by operations personnel to report to appropriate levels of management the necessary information regarding all reactor scrams. A reactor scram is defined as a sudden shutdown of the reactor by the rapid insertion of control rods.

3.3.2 Scram Evaluation - Appendix 12.2

An evaluation used by operations personnel to insure that the causes for unscheduled reactor shutdowns as well as the response of safety-related equipment are fully understood prior to plant restart.

3.3.3 LILCO Deficiency Report (LDR) - see Reference 11.10

1. A report used by operations personnel to report conditions or events which may be reportable to outside agencies as defined in Reference 11.1 or

2. Other nonconformances as defined in Reference 11.10.

3.4 The following topics are contained in this procedure:

		Pages
8.1	Scram Report	3
8.2	Scram Evaluation	3
8.3	LILCO Deficiency Report	4

4.0 PRECAUTIONS

N/A

5.0 PREREQUISITES

N/A

6.0 LIMITATIONS AND ACTIONS

N/A

7.0 MATERIALS AND/OR TEST EQUIPMENT

N/A

8.0 PROCEDURE

8.1 Scram Report

8.1.1 After a scram has occurred, the Watch Engineer or his designee shall prepare a Scram Report.

8.1.2 The Watch Engineer shall review and approve the Scram Report and attachments and forward the report the the Operating Engineer.

8.1.3 The Shift Technical Advisor shall review the report for technical accuracy.

8.1.4 The Operating Engineer shall ensure the Plant Manager, Chief Operating and Technical Engineers, Station Technical Support Manager and OQAE receive a copy of the report.

8.1.5 The Operating Engineer shall ensure charts, recorders etc. are reviewed to determine if any safety limits, trip setpoints, or other limits have been exceeded.

8.1.6 The Operations Section shall ensure the "Scram No." on the Scram Report reflects the total number of scrams to date and ensure cyclic and transient limits of Table 5.7-1 of Technical Specifications are not exceeded.

8.1.7 The Technical Support Manager shall ensure reportability requirements of Reference 11.1 are met.

8.1.8 The PAC shall maintain records associated with all Reactor Scrams.

8.2 Scram Evaluation

8.2.1 After a scram has occurred, the Watch Engineer or his designee shall prepare a scram evaluation.

8.2.2 The Watch Engineer shall review and approve the scram evaluation.

8.2.3 The Shift Technical Advisor shall review the evaluation for technical accuracy.

8.2.4 The Scram Evaluation shall be reviewed with the Operations Engineer prior to commencing a reactor startup.

8.2.5 The Scram Evaluation shall be attached to the Scram Report and forwarded with the report to the Operating Engineer.

8.3 LILCO Deficiency Report

8.3.1 Any station personnel may initiate a LDR in accordance with Reference 11.10.

8.3.2 The initiator should indicate on the LDR whether or not a Technical Specification (TS) or Environmental Technical Specification (ETS) was violated and reference the specification number on the LDR.

8.3.3 The initiator shall immediately inform the On-duty Watch Engineer of all violations of TS or ETS.

8.3.4 The LDR will be processed in accordance with Reference 11.10.

9.0 ACCEPTANCE CRITERIA

N/A

10.0 FINAL CONDITIONS

N/A

11.0 REFERENCES

11.1 SP 12.009.01, Station Reporting Requirements - NRC.

11.2 SP 12.013.01, Maintenance Work Requests.

11.3 SP 12.015.01, Preventive Mantenance Program.

11.4 SP 12.016.01, Surveillance Program.

11.5 SP 21.002.01, Operations Logs and Records.

11.6 Technical Specifications, 9/1/79, Section 6.0

11.7 Regulatory Guide 1.16, Rev. 4, 8/75, Reporting of Operating Information - Appendix A, Technical Specifications.

11.8 Regulatory Guide 10.1, Rev. 3, 5/77, Compilation of Reporting Requirements for Persons Subject to NRC Regulations.

11.9 Environmental Technical Specifications, Rev. 3, Section 5.6.

11.10 QAP-S-15.1, Site OQA Nonconformance Control.

12.0 APPENDICES

12.1 SPF 21.003.01-6, Scram Report.

12.2 SPF 21.003.01-2 , Scram Evaluation.

SCRAM REPORT

SCRAM NUMBER____________________

TIME OF SCRAM___________________

1.0 Mode Switch Position__________________

2.0 Reactor: Critical__________________ Subcritical______________________

3.0 Plant Evolution: Startup_____ Shutting Down _____ Steady Operation _____

4.0 Plant Conditions:

4.1 Reactor Power: __________MWT

4.2 SRMS: A.______cps, B.______cps, C.______cps, D.______cps

4.3 IRM: A.____% range____, D.____% range____, G____% range____

B.____% range____, E____% range____, H____% range____

C.____% range____, F____% range____,

4.4 APRMS: A.____% power, C.____% power, E.____% power,

B.____% power, D.____% power, F.____% power,

4.5 Generator Load ____MWE

4.6 Reactor Pressure: ____psi

4.7 Steam Flow: ____lb/hr.

4.8 Feedwater Flow: ____lb/hr.

4.9 Vessel Water Level: ____

4.10 Recirc System: Loop Manual____, Master Manual ____

4.11 Recirculation Drive Flow: Loop A ______GPM Loop B ______GPM

4.12 Core Flow: ____lb/hr.

5.0 Relief Valve Operation Summary

Valve	Operated	#openings	Manual	Auto	Comments
RV-092A					
RV-092B					
RV-092C					
RV-092D					
RV-092E					
RV-092F					
RV-092G					
RV-092H					
RV-C92J					
RV-092K					
RV-092L					

6.0 External Visual Examination of the suppression chamber completed. (only required if there has been a safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 197°F and reactor coolant system pressure greater than 200 psig.)

7.0 Suppression Chamber drywell vacuum breakers demonstrated operable per SP 24.654.02, if there has been any discharge of steam to the suppression chamber from the safety relief valves.

8.0 Appendix 12.2, Scram Evauluation Report completed and attached.

9.0 Computer sequence of events print out attached.

10.0 A copy of the Watch Engineer's and Control Room Log attached.

11.0 Recorders marked to indicate date and time.

Watch Engineer

Shift Technical Advisor

SCRAM EVALUATION

1.0 Scram Number______ ____________

2.0 Recorder Charts Reviewed

2.1 Reactor Power

2.2 Reactor Level

2.3 Reactor Pressure

2.4 Core Flow

2.5 Feedwater Flow

2.6 Steam Flow

2.7 Station Vent Gas Monitor

3.0 Computer Logs Reviewed

3.1 Sequence of Events Log.

3.2 Alarm Log.

4.0 Equipment Malfunction (include failures of CRD's to insert)

5.0 Procedure Inadequacies

6.0 Operating Errors

7.0 Cause of the Scram

NOTE: R.O.C. must be consulted prior to performing a reactor startup if the cause of the scram can not be positively identified.

8.0 Evaluation to Restart the Reactor

8.1 Has the condition which caused the scram been corrected?

8.2 A sample of reactor coolant has been taken and activity levels are normal.

8.3 All ECCS equipment operated normally consistent with plant conditions.

8.4 All control rods inserted into the core and reactor power decreased at approximately a-90 second period.

8.5 Reactor water level was maintained within normal range consistent with the expected transient.

8.6 Reactor pressure was control normally consistent with the experienced transient.

8.7 All containment parameters were maintained within normal limits consistent with the experienced transient.

8.8 There are no abnormal radiation levels in the plant.

8.9 A determination has been made from the sequence of events recorder that the reactor protection system operated normally.

9.0 Corrective Actions

9.1 Is there any equipment out of service which would prevent the unit from being returned to service?

9.2 Are there any repairs which should be made prior to returning the unit to service?

Report Prepared by:______________________ Date______________ Time______________

Watch Engineer Review____________________ Date______________ Time______________

Shift Technical Advisor Review_________________Date ___________Time ____________

Report reviewed with the Operations Engineer and concurrence received.

Watch Engineer_______________________ Date________________ Time______________

APPENDIX B

SEQUENCE OF EVENTS COMPUTER POINTS VS. ASSOCIATED ALARM POINTS (NSSS/BOP)

July 11, 1983
Rev. 2/JKS

SHOREHAM NUCLEAR POWER STATION - UNIT 1

Comp. Nid Number	Comp. Pt. Id.	Computer Point Description	Assoc. Alarm Number	Annun Window Location	Sys. No.	Annun Pnl Loc.	Alarm Description
C1501	D500	Scram Disch Vol Lvl Ch-A	1203	A9-D3	1C71	603	Discharge Vol Hi Wtr Lvl Trip
C1502	D501	Scram Disch Vol Lvl Ch-B	1203	A9-D3	1C71	603	Discharge Vol Hi Wtr Lvl Trip
C1503	D502	Scram Disch Vol Lvl Ch-C	1187	A10-D3	1C71	603	Discharge Vol Hi Wtr Lvl Trip
C1504	D503	Scram Disch Vol Lvl Ch-D	1187	A10-D3	1C71	603	Discharge Vol Hi Wtr Lvl Trip
C1509	D507	MSIV not full open Ch-A	1204	A9-C4	1C71	603	MSIV Not Full Open Trip
C1510	D508	MSIV not full open Ch-B	1204	A9-C4	1C71	603	MSIV Not Full Open Trip
C1511	D509	MSIV not full open Ch-C	1188	A10-C4	1C71	603	MSIV Not Full Open Trip
C1512	D510	MSIV not full open Ch-D	1188	A10-C4	1C71	603	MSIV Not Full Open Trip
C1513	D511	Contmt Hi Press Ch-A	1205	A9-B1	1C71	603	Pri Cntmt Hi Press Trip
C1514	D512	Contmt Hi Press Ch-B	1205	A9-B1	1C71	603	Pri Cntmt Hi Press Trip
C1515	D513	Contmt Hi Press Ch-C	1189	A10-B1	1C71	603	Pri Cntmt Hi Press Trip
C1516	D514	Contmt Hi Press Ch-D	1189	A10-B1	1C71	603	Pri Cntmt Hi Press Trip
C1517	D515	Reactor Hi Press Ch-A	1206	A9-B2	1C71	603	Rx Vessel Hi Press Trip
C1518	D516	Reactor Hi Press Ch-B	1206	A9-B2	1C71	603	Rx Vessel Hi Press Trip
C1519	D517	Reactor Hi Press Ch-C	1190	A10-B2	1C71	603	Rx Vessel Hi Press Trip
C1520	D518	Reactor Hi Press Ch-D	1190	A10-B2	1C71	603	Rx Vessel Hi Press Trip
C1521	D519	Reactor Lo Wtr Lvl Ch-A	1207	A9-B3	1C71	603	Rx Vessel Lo Lvl Trip
C1522	D520	Reactor Lo Wtr Lvl Ch-B	1207	A9-B3	1C71	603	Rx Vessel Lo Lvl Trip
C1523	D521	Reactor Lo Wtr Lvl Ch-C	1191	A10-B3	1C71	603	Rx Vessel Lo Lvl Trip
C1524	D522	Reactor Lo Wtr Lvl Ch-D	1191	A10-B3	1C71	603	Rx Vessel Lo Lvl Trip
C1525	D523	Main Steam Line Hi Rad A	1208	A9-B4	1C71	603	Mn Stm Line Hi Rad Trip
C1526	D524	Main Steam Line Hi Rad B	1208	A9-B4	1C71	603	Mn Stm Line Hi Rad Trip
C1527	D525	Main Steam Line Hi Rad C	1192	A10-B4	1C71	603	Mn Stm Line Hi Rad Trip
C1528	D526	Main Steam Line Hi Rad D	1192	A10-B4	1C71	603	Mn Stm Line Hi Rad Trip
C1529	D527	Neutron Monitoring Ch-A1	1212	A9-C3	1C71	603	Neutron Monitoring Sys Trip
C1530	D528	Neutron Monitoring Ch-B1	1212	A9-C3	1C71	603	Neutron Monitoring Sys Trip
C1531	D529	Neutron Monitoring Ch-A2	1196	A10-C3	1C71	603	Neutron Monitoring Sys Trip
C1532	D530	Neutron Monitoring Ch-B2	1196	A10-C3	1C71	603	Neutron Monitoring Sys Trip
C1535	D566	RX Man Scram Ch-C	1216	A10-A3	1C71	603	Rx Man Scram Sys B
C1536	D567	RX Man Scram Ch-D	1216	A10-A3	1C71	603	Rx Man Scram Sys B
C1537	D533	RX Man Scram Ch-A	1215	A9-A3	1C71	603	Rx Man Scram Sys A
C1538	D534	RX Man Scram Ch-B	1215	A9-A3	1C71	603	Rx Man Scram Sys A
C1539	D535	RX Auto Scram Ch-A	1213	A9-A2	1C71	603	Rx Auto Trip Ch A2
C1540	D536	RX Auto Scram Ch-B	1214	A10-A2	1C71	603	Rx Auto Trip Ch B2
C1552	D537	NSS Spare Sequence Annun			1C71		
C1553	D538	TSV Closure Ch-A	1211	A9-D1	1C71	603	Turb Stop Vv Closure Trip
C1554	D539	TSV Closure Ch-B	1211	A9-D1	1C71	603	Turb Stop Vv Closure Trip
C1555	D540	TSV Closure Ch-C	1195	A10-D1	1C71	603	Turb Stop Vv Closure Trip
C1556	D541	TSV Closure Ch-D	1195	A10-D1	1C71	603	Turb Stop Vv Closure Trip
C1557	D542	TCV Fast Close Ch-A	1209	A9-D2	1C71	603	Turb Control Vv Fast Close Trip
C1558	D543	TCV Fast Close Ch-B	1209	A9-D2	1C71	603	Turb Control Vv Fast Close Trip
C1559	D544	TCV Fast Close Ch-C	1193	A10-D2	1C71	603	Turb Control Vv Fast Close Trip
C1560	D545	TCV Fast Close Ch-D	1193	A10-D2	1C71	603	Turb Control Vv Fast Close Trip
C1561	D546	APRM Ch-A Upscale Lvl	1228	A10-A9	1C51	603	APRM Bus A Upscl Trip or Inop
C1562	D547	APRM Ch-B Upscale Lvl	1229	A10-A10	1C51	603	APRM Bus B Upscl Trip or Inop

SEQUENCE OF EVENTS COMPUTER POINTS VS. ASSOCIATED ALARM POINTS (NSSS/BOP)

SHOREHAM NUCLEAR POWER STATION - UNIT 1

Comp. Nid Number	Comp. Pt. Id.	Computer Point Description	Assoc. Alarm Number	Annun Window Location	Sys. No.	Annun Pnl Loc.	Alarm Description
C1563	D548	APRM Ch-C Upscale Lvl	1228	A10-A9	1C51	603	APRM Bus A Upscl Trip or Inop
C1564	D549	APRM Ch-D Upscale Lvl	1229	A10-A10	1C51	603	APRM Bus B Upscl Trip or Inop
C1565	D550	APRM Ch-E Upscale Lvl	1228	A10-A9	1C51	603	APRM Bus A Upscl Trip or Inop
C1566	D551	APRM Ch-F Upscale Lvl	1229	A10-A10	1C51	603	APRM Bus B Upscl Trip or Inop
C1567	D552	NSS Spare Sequence Annun					
C1568	D553	NSS Spare Sequence Annun					
C1569	D554	IRM Ch-A Upscale Lvl	1239	A10-A7	1C51	603	IRM Trip Sys A Upscl/Inop
C1570	D555	IRM Ch-B Upscale Lvl	1241	A10-A8	1C51	603	IRM Trip Sys B Upscl/Inop
C1571	D556	IRM Ch-C Upscale Lvl	1239	A10-A7	1C51	603	IRM Trip Sys A Upscl/Inop
C1572	D557	IRM Ch-D Upscale Lvl	1241	A10-A8	1C51	603	IRM Trip Sys B Upscl/Inop
C1573	D558	IRM Ch-E Upscale Lvl	1239	A10-A7	1C51	603	IRM Trip Sys A Upscl/Inop
C1574	D559	IRM Ch-F Upscale Lvl	1241	A10-A8	1C51	603	IRM Trip Sys B Upscl/Inop
C1575	D560	IRM Ch-G Upscale Lvl	1239	A10-A7	1C51	603	IRM Trip Sys A Upscl/Inop
C1576	D561	IRM Ch-H Upscale Lvl	1241	A10-A8	1C51	603	IRM Trip Sys B Upscl/Inop
C1579	D562	NSS Spare Sequence Annun					
C1584	D563	NSS Spare Sequence Annun					
C1591	D564	NSS Spare Sequence Annun					
C1592	D565	NSS Spare Sequence Annun					
C1756	D626	APRM Thermal Lvl Trip A	1228	A10-A9	1C51	603	APRM Bus A Upscl Trip or Inop
C1757	D627	APRM Thermal Lvl Trip B	1229	A10-A10	1C51	603	APRM Bus B Upscl Trip or Inop
C1758	D628	APRM Thermal Lvl Trip C	1228	A10-A9	1C51	603	APRM Bus A Upscl Trip or Inop
C1759	D629	APRM Thermal Lvl Trip D	1229	A10-A10	1C51	603	APRM Bus B Upscl Trip or Inop
C1760	D630	APRM Thermal Lvl Trip E	1228	A10-A9	1C51	603	APRM Bus A Upscl Trip or Inop
C1761	D631	APRM Thermal Lvl Trip F	1229	A10-A10	1C51	603	APRM Bus B Upscl Trip or Inop
C3801	D140	Mn Turb Exh Hood Hi Temp	0142	209F-E8	1N32	B	Main Turb Exh Hood Hi Temp Trip
C3802	D141	Mn Turb Stator Clg Loss	0132	209F-C10	1N32	B	Main Turb Stator Clg Loss Trip
C3803	D142	Turb Shaft Lo PP Lo P	0129	209F-E7	1N32	B	Main Turb Shaft Pump Lo Press Trip
C3804	D143	Mn Turb Thrust Brg Wear	0134	209F-C8	1N32	B	Main Turb Thrust Brg Wear Trip
C3805	D144	Mn Turb 125V DC Trip	0144	209F-D9	1N32	B	Main Turb EHC 125V DC Trip Loss
C3806	D145	EHC Loss of Fluid	0141	209F-D8	1N32	B	EHC Loss of Fluid Trip
C3807	D146	MN Turb No Speed Signal	0138	209F-B10	1N32	B	Main Turb No Speed Signal Trip
C3808	D147	Mn Turb Vib Hi	0143	209F-D10	1N32	B	Main Turb vib Hi Trip
C3809	D148	Mn Turb Lo Cond Vac	0130	209F-E9	1N32	B	Main Turb Cond Lo Vac Trip
C3810	D149	Loss of 24V DC Pwr	0140	209F-A9	1N32	B	Main Turb Tripped
C3811	D150	Mn Turb Man Trip	0139	209F-B8	1N32	B	Main Turb Manual Trip
C3812	D151	Mn Turb Tripped	0140	209F-A9	1N32	B	Main Turb Tripped
C3813	D152	Mn Turb Backup Overspeed	0137	209F-B9	1N32	B	Main Turb Backup Overspeed Trip
C3814	D153	Mn Turb Msr Drn Tk Hi	0135	209F-C9	1N32	B	Main Turb Msr Drn Tk Level Hi Trip
C3816	D155	Loss of Emerg Trip Fluid	0140	209F-A9	1N32	B	Main Turb Tripped
C3832	D159	BOP Spare Sequence Annun					
C3833	D160	BOP Spare Sequence Annun					
C3834	D161	BOP Spare Sequence Annun					
C5417	D156	Loss of Circ Wtr	0140	209F-A9	1N71	B	Main Turb Tripped
C5433	D154	Loss of Circ Wtr	0140	209F-A9	1N71	B	Main Turb Tripped
C7720	D101	Volts/Hz GT 110 P/C Prot	0236	209G-C5	1R61	B	Main Gen Volts per Hertz Hi

SEQUENCE OF EVENTS COMPUTER POINTS VS. ASSOCIATED ALARM POINTS (NSSS/BOP)

SHOREHAM NUCLEAR POWER STATION - UNIT 1

Comp. Nid Number	Comp. Pt. Id.	Computer Point Description	Assoc. Alarm Number	Annun Window Location	Sys. No.	Annun Pnl Loc.	Alarm Description
C7721	D102	Volts/Hz GT 118 P/C Prot	0236	209G-C5	1R61	B	Main Gen Volts per Hertz HI
C7722	D117	Unit Diff PRI Prot	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7723	D118	Gen Neut Grd Pri Prot	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7724	D119	Max Exc Lim Unit Pri Prot	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7725	D120	MN XFMR 1A Sudden P	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7726	D121	MN XFMR 1B Sudden P	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7727	D122	MN Gen Line Diff Prot	0147	209G-04	1R61	B	Main Gen Prot Lctl
C7728	D123	Unit BKR 1350/1360 Fail	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7729	D124	Unit Antimotoring Prot	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7730	D125	MN XFMR 1A Diff B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7731	D126	MN XFMR 1B Diff B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7732	D127	MN Gen Diff B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7733	D128	Exc-Alt Diff B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7734	D129	Gen Neut Grd B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7735	D130	Gen Grd Bus B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7736	D131	Loss of Exc B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7737	D132	Gen Rev Pwr B/U Prot	0213	209G-A6	1R61	B	Unit Backup Prot Trip
C7738	D135	MN XFMR 1A Grd B/U Prot	0214	209G-B6	1R61	B	Sys Backup Prot Trip
C7739	D136	MN XFMR 1B Grd B/U Prot	0214	209G-B6	1R61	B	Sys Backup Prot Trip
C7740	D137	Neg Phase Seq B/U Prot	0214	209G-B6	1R61	B	Sys Backup Prot Trip
C7741	D138	Offset MHO Rly B/U Prot	0214	209G-B6	1R61	B	Sys Backup Prot Trip
C7742	D139	Gen Fld Grd PRI Prot	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7743	D157	Unit Gen Leads OC Prot	0212	209G-A5	1R61	B	Unit Pri Prot Trip
C7760	D162	Bop Spare Sequence Annun					
C7761	D163	Bop Spare Sequence Annun					
C7762	D164	Bop Spare Sequence Annun					
C7763	D165	Bop Spare Sequence Annun					
C7821	D166	Bop Spare Sequence Annun					
C7822	D167	Bop Spare Sequence Annun					
C7823	D103	NSS B/U OC Prot	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7824	D104	NSS B/U GRD OC x WDG	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7825	D105	NSS B/U GRD OC Y WDG	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7826	D106	NSS Xfmr Sudden Press	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7827	D107	NSS Xfmr Diff Pri Prot	0218	209H-A1	1R62	B	NSS Xfmr Pri Prot Trip
C7828	D108	NSS Bkr 1350/1360 Fail	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7829	D109	RSS Xfmr Diff Pri Prot	0220	209H-A6	1R62	B	RSS Xfmr Pri Prot Trip
C7830	D110	RSS B/U OC Prot	0221	209H-A5	1R62	B	RSS Xfmr Backup Prot Trip
C7831	D111	RSS B/U Grd OC x WDG	0221	209H-A5	1R62	B	RSS Xfmr Backup Prot Trip
C7832	D112	RSS B/U Grd OC Y WDG	0221	209H-A5	1R62	B	RSS Xfmr Backup Prot Trip
C7833	D113	RSS Xfmr Sudden Press	0221	209H-A5	1R62	B	RSS Xfmr Backup Prot Trip
C7834	D114	RSS Xfmr 69KV Bkr B/U	0221	209H-A5	1R62	B	RSS Xfmr Backup Prot Trip
C7835	D115	RSS Xfmr 69KV Bus Diff	0220	209H-A6	1R62	B	RSS Xfmr Pri Prot Trip
C7836	D116	13KV Gas Turb Bkr Fail	0221	209H-A5	1R62	B	RSS Xfmr Backup Prot Trip
C7837	D133	Gen Bkr 1310 B/U Prot	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7838	D134	Gen Bkr 1330 B/U Prot	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip

Comp. Nid Number	Comp. Pt. Id.	Computer Point Description	Assoc. Alarm Number	Annun Window Location	Sys. No.	Annun Pnl Loc.	Alarm Description
C7841	D158	NSS Gen Leads OC Prot	0219	209H-A2	1R62	B	NSS Xfmr Backup Prot Trip
C7842	D168	Bop Spare Sequence Annun					
C7843	D169	Bop Spare Sequence Annun					

APPENDIX C

12-08-83

```
135658 ALM  A543 ROD OUT BLOCK             ON
135743 NORM A528 APRM UPSCL                NORM
135743 NORM A543 ROD OUT BLOCK             OFF
135815 ALM  A528 APRM UPSCL                ALM
135815 ALM  A543 ROD OUT BLOCK             ON
       140347   44  SEQ D547 APRM CHNL B UPSCALE LVL  TRIP
       140347   44  SEQ D528 NEUTRON MONITORING CH B1 TRIP
       140347   44 *SEQ D529 NEUTRON MONITORING CH A2 TRIP
       140347   44  SEQ D536 RX AUTO SCRAM CHNL B     TRIP
       140552   35  SEQ D528 NEUTRON MONITORING CH B1 RSET
       140552   35 *SEQ D529 NEUTRON MONITORING CH A2 RSET
       140552   35 *SEQ D547 APRM CHNL B UPSCALE LVL  RSET
140553 ALM  A537 APRM BYPASSED CH B        ON
140553 NORM A528 APRM UPSCL                NORM
140553 NORM A543 ROD OUT BLOCK             OFF
       140555   57  SEQ D536 RX AUTO SCRAM CHNL B     RSET
140836 NORM A537 APRM BYPASSED CH B        OFF
140923 ALM  E127 EMER BUS 101 DG PRI PROT  TRIP
140927 ALM  A537 APRM BYPASSED CH B        ON
140938 NORM E127 EMER BUS 101 DG PRI PROT  NORM
140939 NORM A537 APRM BYPASSED CH B        OFF
140951 NORM E237 DG 103 LUBE OIL HTR       NORM
141002 NORM B001 APRM B % POWER                       -0.4
141203 ALM  E127 EMER BUS 101 DG PRI PROT  TRIP
141204 ALM  A539 APRM BYPASSED CH D        ON
141215 NORM E127 EMER BUS 101 DG PRI PROT  NORM
141245 ALM  E127 EMER BUS 101 DG PRI PROT  TRIP
141250 NORM E127 EMER BUS 101 DG PRI PROT  NORM
141310 ALM  E127 EMER BUS 101 DG PRI PROT  TRIP
141321 NORM E127 EMER BUS 101 DG PRI PROT  NORM
141630 ALM  E236 DG 102 LUBE OIL HTR       LCTL
141631 ALM  E127 EMER BUS 101 DG PRI PROT  TRIP
141703 LRL  B003 APRM D % POWER
141717 NORM A539 APRM BYPASSED CH D        OFF
141824 ALM  A528 APRM UPSCL                ALM
141824 ALM  A543 ROD OUT BLOCK             ON
       142542   32  SEQ D549 APRM CHNL D UPSCALE LVL  TRIP
       142542   33  SEQ D530 NEUTRON MONITORING CH B2 TRIP
       142542   33  SEQ D536 RX AUTO SCRAM CHNL B     TRIP
       142624   49  SEQ D530 NEUTRON MONITORING CH B2 RSET
       142624   49 *SEQ D549 APRM CHNL D UPSCALE LVL  RSET
142625 ALM  A539 APRM BYPASSED CH D        ON
       142625   55  SEQ D536 RX AUTO SCRAM CHNL B     RSET
142625 NORM A528 APRM UPSCL                NORM
142625 NORM A543 ROD OUT BLOCK             OFF
142629 ALM  E233 DG 103 AIR CPRSR 3C       LCTL
142629 ALM  E234 DG 103 AIR CPRSR 4C       LCTL
142902 NORM B003 APRM D % POWER                       -0.6
142903 NORM A539 APRM BYPASSED CH D        OFF
142956 ALM  A541 APRM BYPASSED CH F        ON
143303 LRL  B005 APRM F % POWER
143305 NORM A541 APRM BYPASSED CH F        OFF
143309 NORM E127 EMER BUS 101 DG PRI PROT  NORM
143406 ALM  A528 APRM UPSCL                ALM
143406 ALM  A543 ROD OUT BLOCK             ON
143452 NORM A528 APRM UPSCL                NORM
143452 NORM A543 ROD OUT BLOCK             OFF
143508 ALM  A528 APRM UPSCL                ALM
```

SEQUENCE OF EVENTS (SOE)

SOE FORMAT

TIME	CYCLE	PT. ID.	NAME	STATUS
xxxxxx	xx	xxxx	24 Characters	xxxx

SEQUENCE OF EVENTS (SOE)

APPENDIX D

Process Computer Post Trip Log

Analog Parameters

A. NSSS Log

1. APRM A
2. APRM B
3. Reactor Pressure
4. Core Differential Pressure
5. Jet Pump Total Flow
6. Feedwater Flow A
7. Feedwater Flow B
8. Reactor Water Level
9. Total Steam Flow
10. Feedwater Temperature A

B. BOP Log

1. Generator Current Phase A
2. Generator Voltage
3. Condenser Pressure A
4. Condenser Pressure B
5. Condensate Booster Pump Discharge Pressure
6. Condensate Pump Discharge Header Pressure
7. Condensate Booster Pump A Flow
8. Condensate Booster Pump B Flow
9. Feedwater Pump A Suction Flow
10. Feedwater Pump B Suction Flow
11. Feedwater Pump A Discharge Pressure
12. Feedwater Pump B Discharge Pressure
13. Generator Stator Cooling Water Temperature
14. Generator Field Current
15. Generator Field Voltage
16. Drywell Pressure
17. Low Pressure Turbine A Exhaust Hood Temperature
18. Low Pressure Turbine B Exhaust Hood Temperature
19. Turbine Shaft Driven Oil Pump Discharge Pressure
20. Turbine Bearing Oil Supply Pressure
21. MSR Drain Tank A Level
22. MSR Drain Tank B Level
23. Turbine Thrust Bearing Wear
24. thru 31. Main Turbine Bearing Vibration (8 Points)
32. and 33. Exciter Bearing Vibration (2 Points)
34. Turbine Speed
35. Spare
36. Condensate Pump A Flow (Calculated Valve)
37. Condensate Pump B Flow (Calculated Valve)
38. Unit Gross Power (Calculated Valve)
39. Normal Station Service Transformer Power (Calculated Valve)
40. Reserve Station Service Transformer Power (Calculated Valve)

APPENDIX E

ERF PHASE I DATA SET AVAILABL FROM PROCESS COMPUTER

	PARAMETER	INSTRUMENT
1.	Reactor Pressure	1B21*PT004A
2.	Reactor Pressure	1B21*PT004B
3.	Reactor Wtr Lev (WR)	1B21*LIT004A
4.	Reactor Wtr Lev (WR)	1B21*LIT004B
5.	Reactor Wtr Lev (FZ)	1B21*LIT007A
6.	Reactor Wtr Lev (FZ)	1B21*LIT007B
7.	ADS/SRV Tailpipe Press	1B21*PT153A
8.	ADS/SRV Tailpipe Press	1B21*PT153B
9.	ADS/SRV Tailpipe Press	1B21*PT153C
10.	ADS/SRV Tailpipe Press	1B21*PT153D
11.	ADS/SRV Tailpipe Press	1B21*PT153E
12.	ADS/SRV Tailpipe Press	1B21*PT153F
13.	ADS/SRV Tailpipe Press	1B21*PT153G
14.	ADS/SRV Tailpipe Press	1B21*PT153H
15.	ADS/SRV Tailpipe Press	1B21*PT153J
16.	ADS/SRV Tailpipe Press	1B21*PT153K
17.	ADS/SRV Tailpipe Press	1B21*PT153L
18.	RCIC Pump Disch Flow	1E51*FT003
19.	RHR Sys A Flow	1E11*FT001A
20.	RHR Sys B Flow	1E11*FT001B
21.	RHR HX A Outlet Temp	1E11*TE012A
22.	RHR HX B Outlet Temp	1E11*TE012B
23.	RHR HX A Inlet Temp	1E11*TE011A
24.	RHR HX B Inlet Temp	1E11*TE011B
25.	HPCI Pump Disch Flow	1E41*FT003
26.	Core Spray Sys A Flow	1E21*FT002A
27.	Core Spray Sys B Flow	1E21*FT002B
28.	RHR HX A - SW OUtlet Temp	1E11*TE013A
29.	RHR HX B - SW Outlet Temp	1E11*TE013B
30.	RHR Svce Wtr A Flow	1E11*FT006A
31.	RHR Svce Wtr B Flow	1E11*FT006B
32.	Reactor Bldg. Flood Level	1G11*LTS645A
33.	Reactor Bldg. Flood Level	1G11*LTS645B
34.	Drywell Pressure	1Z93*PT003A
35.	Drywell Pressure	1Z93*PT003B
36.	Suppression Chamber Press	1Z93*PT004A
37.	Suppression Chamber Press	1Z93*PT004B
38.	Suppression Pool Wtr Temp (1 ft)	1Z93*TE110Z
39.	Suppression Pool Wtr Temp (1 ft)	1Z93*TE111W
40.	Suppression Pool Wtr Temp (1 ft)	1Z93*TE112Y
41.	Suppression Pool Wtr Temp (1 ft)	1Z93*TE113X

	PARAMETER	INSTRUMENT
42.	Suppression Pool Wtr Temp (2 ft)	1Z93*TE132A
43.	Suppression Pool Wtr Temp (2 ft)	1Z93*TE133B
44.	Suppression Pool Wtr Temp (2 ft)	1Z93*TE134A
45.	Suppression Pool Wtr Temp (2 ft)	1Z93*TE135B
46.	Suppression Pool Wtr Level	1Z93*LT001A
47.	Suppression Pool Wtr Level	1Z93*LT001B
48.	Drywell Hydrogen Conc.	1T48*H2Z115A
49.	Drywell Hydrogen Conc.	1T48*H2Z115B
50.	Suppression Chamber H_2 Conc	1T48*H2Z116A
51.	Suppression Chamber H_2 Conc	1T48*H2Z116B
52.	Drywell Oxygen Conc	1T48*O2Z123A
53.	Drywell Oxygen Conc	1T48*O2Z123B
54.	Suppression Chamber O_2 Conc	1T48*O2Z124A
55.	Suppression Chamber O_2 Conc	1T48*O2Z124B
56.	Reactor Bldg. Press.	1T41-PDT011
57.	ADS/SRV Air Hdr. A Press	1P50*PT116A
58.	ADS/SRV Air Hdr. B Press	1P50*PT116B
59.	Drywell Temperature	1T47TE027A
60.		
61.		
62.		
63.		
64.		
65.		
66.		
67.		
68.		
69.	Drywell Temperature	1T47TE027L
70.	RBCLCW HX A Outlet Temp	1P42-TE001A
71.	RBCLCW HX B Outlet Temp	1P42-TE001B
72.	Circ Wtr Pmp A Disch Press	1N71-PT083A
73.	Circ Wtr Pmp B Disch Press	1N71-PT083B
74.	Circ Wtr Pmp C Disch Press	1N71-PT083C
75.	Circ Wtr Pmp D Disch Press	1N71-PT083D
76.	Main Condenser Pressure	1N21-PT005A
77.	Main Condenser Pressure	1N21-PT005B
78.	Condensate Storage Tk. Level	1P11-LT002
79.	Feedwater Temperature	1B21-TT001A
80.	Feedwater Temperature	1B21-TT001B
81.	Feedwater Temperature	1B21-TT001C
82.	Feedwater Temperature	1B21-TT001D
83.	Feedwater Flow	1C32-FT001A
84.	Feedwater Flow	1C32-FT001B
85.	Neutron Flux Level	APRM A
86.	Neutron Flux Level	APRM B
87.	Neutron Flux Level	APRM C
88.	Neutron Flux Level	APRM D
89.	Neutron Flux Level	APRM E
90.	Neutron Flux Level	APRM F
91.	Neutron Flux Level	TIP A
92.	Neutron Flux Level	TIP B
93.	Neutron Flux Level	TIP C
94.	Neutron Flux Level	TIP D
95.	Control Rod Position (Core Map Graphic Display)	

y-11600.02-758c 12/13/82 033

	CHANNEL ID	DESCRIPTION	DATA INDEX	RANGE	ALARM LIMITS	TECH SPEC LIMITS
1.	AM01	RHR PUMPS WEST	1(G)	.1-1000MREM/HR	5MREM/HR	
2.	AM02	HPCI TURBINE	1(G)	.1-1000MREM/HR	5MREM/HR	
3.	AM03	RHR PUMPS EAST	1(G)	.1-1000MREM/HR	5MREM/HR	
4.	AM04	RHR HEAT EXC WEST	1(G)	.1-1000MREM/HR	5MREM/HR	
5.	AM05	RHR HEAT EXC EAST	1(G)	.1-1000MREM/HR	5MREM/HR	
6.	AM06	P&V AREA WEST	1(G)	$1-10^4$MREM/HR		
7.	AM07	P&V AREA EAST	1(G)	$1-10^4$MREM/HR		
8.	AM08	PERSONNEL HATCH	1(G)	.1-1000MREM/HR	5MREM/HR	
9.	AM09	EQUIP HATCH	1(G)	.1-1000MREM/HR	5MREM/HR	
10.	AM10	FUEL POOL CLEAN UP PUMPS	1(G)	.1-100MREM/HR	5MREM/HR	
11.	AM11	RW CLEAN UP PUMPS	1(G)	.1-100MREM/HR		
12.	AM12	FUEL POOL HEAT EXC	1(G)	.1-1000MREM/HR		
13.	AM13	CONTAM EQUIPMENT	1(G)	.1-1000MREM/HR	5MREM/HR	
14.	AM14	FUEL POOL	1(G)	.1-1000MREM/HR	5MREM/HR	
15.	AM15	REA HEAT INSUL STG	1(G)	.1-1000MREM/HR	5MREM/HR	
16.	AM16	DECON AREA WEST	1(G)	.1-1000MREM/HR	5MREM/HR	
17.	AM17	CONDENSATE PUMPS	1(G)	.1-1000MREM/HR	5MREM/HR	
18.	AM18	DECON AREA EAST	1(G)	.1-1000MREM/HR		
19.	AM19	HI PRESS TURBINE	1(G)	.1-1000MREM/HR		
20.	AM20	HOIST AREA	1(G)	.1-1000MREM/HR		
21.	AM21	DECON AREA	1(G)	.1-1000MREM/HR		
22.	AM22	CHEM LAB	1(G)	.01-100MREM/HR	1MREM/HR	
23.	AM23	HEATER BAYS	1(G)	.1-1000MREM/HR	5MREM/HR	
24.	AM24	DECON AREA NORTH	1(G)	.1-1000MREM/HR	5MREM/HR	
25.	AM25	WASTE EVAP	1(G)	.1-1000MREM/HR	5MREM/HR	

	CHANNEL ID	DESCRIPTION	DATA INDEX	RANGE	ALARM LIMITS	TECH SPEC LIMITS	
26.	AM26	SOLID WASTE CASK AREA	1(G)	1-100 MREM/HR			1.51
27.	AM27	RADW SAMPLE ROOM	1(G)	.1-1000MREM/HR			1.52
28.	AM28	REGENERANT EVAP	1(G)	.1-1000MREM/HR			1.53
29.	AM29	RE/ÆN FEED EVAP H2O TANK	1(G)	.1-1000MREM/HR			1.54
30.	AM30	CONTROL ROOM	1(G)	.01-100MREM/HR	1MRE?/HR		1.55
31.	AM32	REGENERANT EVAP AREA	1(G)	.1-10^{4}MREM/HR			1.56
32.	AM33	HOIST AREA	1(G)	.1-10^{4}MREM\HR	5MREM\HR		1.57
33.	AM34	DEMINERALIZER AREA	1(G)	.1-10^{4}MREM//HR			1.58
34.	AM35	DRYWELL EQUIP DRAIN	1(G)	.1-10^{4}MREM/HR	5MREM/HR		1.59
35.	AM36	TIP DRIVE ROOM	1(G)	1-10^{4}MREM/HR	5MREM/HR		1.60
36.	AM37	TIP DRIVE AREA	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.1
37.	AM38	NEW FUEL VAULT AREA	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.2
38.	AM39	CORRIDOR AREA	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.3
39.	AM40	RAMP AREA	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.4
40.	AM41	HOIST AREA	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.?
41.	AM42	PUMP AREA	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.6
42.	AM43	SAMPLE ROOM AREA MONITOR	1(G)	.1-10^{4}MREM/HR	5MREM/HR		2.7
43.	AM44	FLASK AREA MONITOR	1(G)	.1-10^{4}MREM/HR			2.8
44.	AM45	EQUIPMENT ROOM AREA MON	1(G)	.1-10^{4}MREM/HR			2.9
45.	AM85A	H.R. CONTAINMENT AREA A	1(G)	1-10^{7}REM/HR			2.10
46.	AM85B	H.R. CONTAINMENT AREA B	1(G)	1-10^{7}REM/HR			2.11
47.	FLOW1	OFFGAS SYS FLOW	1(F)	(LTR)			2.12
48.	MET1	WIND SPEED 150	1(WS)	0-100MPH			2.13
49.	MET2	WIND SPEED 33	1(WS)	0-100MPH			2.14
50.	MET3	WIND DIRECTION 150	1(WD)	0-540DEGREES			2.15

	CHANNEL ID	DESCRIPTION	DATA INDEX	RANGE	ALARM LIMITS	TECH SPEC LIMITS	
51.	MET4	WIND DIRECTION 33	1(WD)	0-540 DEGREES			2.16
52.	MET5	DELTA TEMP 33 TO 150	1(T)	-10°F-20°F			2.17
53.	MET6	TEMPERATURE AT 33	1(T)	-25°F-125°F			2.18
54.	PM11A	MAIN STEAM LINE A	1(G)	1-10[illegible]MREM/HR			2.19
55.	PM11B	MAIN STEAM LINE B	1(G)	1-10[illegible]MREM/HR			2.20
56.	PM11C	MAIN STEAM LINE C	1(G)	1-10[illegible]MREM/HR			2.21
57.	PM11D	MAIN STEAM LINE D	1(G)	1-10[illegible]MREM/HR			2.22
58.	PM126	HIGH RANGE STC VNT EXH	1(G)	1-10[illegible]MC/CC			2.23
59.	PM127	PAS VNT MONITOR PART	1(G)	10-[illegible]-10-[illegible]MC/CC			2.24
60.	PM128	PAS VNT MONITOR GAS	1(G)	2X10-[illegible]-2X10[illegible] MCC/CC			2.25
61.	PM12A	STEAM JET AIR EJE A	1(G)	1-10[illegible]MREM/HR			2.26
62.	PM12B	STEAM JET AIR EJE B	1(G)	1-10[illegible]MREM/HR			2.27
63.	PM13	LIQ RADWASTE DISCHARGE	1(G)	1X10-[illegible]-1X10-[illegible]MCC/CC			2.28
64.	PM13	LIQ RADWASTE DISCHARGE	4(F)	0-20CGPM			2.29
65.	PM134	HIGH RANGE RBSYS	1(G)	1-10[illegible]MC/CC			2.30
66.	PM14	STEAM JET AIR EJE C	1(G)	1-10[illegible]MREM/HR			2.31
67.	PM17A	REFUEL LEVEL VENT A	1(G)	.01-100MREM/HR			2.32
68.	PM17B	REFUEL LEVEL VENT B	1(G)	.01-100MREM/HR			2.33
69.	PM19	CONT DRY FILT TRAIN EX	1(G)	10-[illegible]-10-[illegible]MC/CC			2.34
70.	PM21	REACTOR BLDG SBVENT 1	1(G)	10-[illegible]-10[illegible]MC/CC			2.35
71.	PM21	REACTOR BLDG SBVENT 1	4(F)	0-4CFM			2.36
72.	PM22	REACTOR BLDG SBVENT 2	1(G)	10-[illegible]-10[illegible]MC/CC			2.37
73.	PM22	REACTOR BLDG SBVENT 2	4(F)	0-4 CFM			2.38
74.	PM23A	RHR HEAT EXC OUTLET A	1(G)	10-[illegible]-10-[illegible]MC/CC			2.39
75.	PM23B	RHR HEAT EXC OUTLET B	1(G)	10-[illegible]-10-[illegible] MC/CC			2.40

	CHANNEL ID	DESCRIPTION	DATA INDEX	RANGE	ALARM LIMITS	TECH SPEC LIMITS	
76.	PM24	REA BLDG CLO LOOP WATER	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.41
77.	PM25A	CONT ROOM VENT 1	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.42
78.	PM25B	CONT ROOM VENT 2	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.43
79.	PM26A	CONT ROOM VENT 3	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$/MC/CC			2.44
80.	PM26B	CONT ROOM VENT 4	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.45
81.	PM27	CONT ROOM ATMOS PART	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.46
82.	PM28	CONT ROOM ATMOS GAS	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.47
83.	PM29	REA BLDG VENT EX GAS	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.48
84.	PM30	REA BLDG VENT EX PART	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.49
85.	PM41	STATION VENT EX PART	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.50
86.	PM41	STATION VENT EX PART	4(F)	0-4CFM			2.51
87.	PM42	STATION VENT EX GAS	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.52
88.	PM42	STATION VENT EX GAS	4(F)	0-4CFM			2.53
89.	PM43	STATION VENT EX IOD	1(N)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.54
90.	PM43	STATION VENT EX IOD	4(F)	0-4 CFM			2.55
91.	PM45	RADWASTE STEAM GEN N16	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			2.56
92.	PM46	WASTE EVAPORATOR	1(G)	$10^{[illegible]}$-$10^{[illegible]}$MREM/HR			2.57
93.	PM47	REGENERATE EVAPORATOR	1(G)	$10^{[illegible]}$-$10^{[illegible]}$MREM/HR			2.58
94.	PM48	SPENT RESIN TRANS TANK	1(G)	$10^{[illegible]}$-$10^{[illegible]}$MREM/HR			2.59
95.	PM49	EVAP BOTTOMS TANK	1(G)	$10^{[illegible]}$-$10^{[illegible]}$MREM/HR			2.60
96.	PM51	AIR REM PUMP DISCHARGE	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			3.1
97.	PM52	HOT WATER HEAT SYS & EXC	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			3.2
98.	PM53	STEAM SEAL EVAPORATOR	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$ MC/CC			3.3
99.	PM54	TUR BLDG CLO LOOP WATER	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC			3.4
100.	PM55	RAD BLDG VENT EX GAS	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$ MC/CC			3.5

	CHANNEL ID	DESCRIPTION	DATA INDEX	RANGE	ALARM LIMITS	TECH SPEC LIMITS
101.	PM56	RAD BLDG VENT EX PART	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.6
102.	PM57	TUR BLDG VENT EX GAS	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.7
103.	PM58	TUR BLDG VENT EX PART	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.8
104.	PM61	CONT DRYWELL PART	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.9
105.	PM62	CONT DRYWELL GAS	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.10
106.	PM65A	OFFGAS SYS DISCH A	1(G)	.01-100MREM/HR		3.11
107.	PM65B	OFFGAS SYS DISCH B	1(G)	.01-100MREM/HR		3.12
108.	PM66	REACTOR BLDG PAM	1(G)	$10^{-[illegible]}$-$10^{[illegible]}$ MC/CC		3.13
109.	PM66	REACTOR BLDG PAM	4(F)	(LTR)		3.14
110.	PM67	TURBINE BLDG PAM	1(G)	$10^{-[illegible]}$-$10^{[illegible]}$MC/CC		3.15
111.	PM67	TURBINE BLDG PAM	4(F)	(LTR)		3.16
112.	PM68	RADWASTE BLDG PAM	1(G)	$10^{-[illegible]}$-$10^{[illegible]}$MC/CC		3.17
113.	PM68	RADWASTE BLDG PM	4(F)	(LTR)		3.18
114.	PM69	STATION VENT PAM	1(G)	$10^{-[illegible]}$-$10^{[illegible]}$MC/CC		3.19
115.	PM69	STATION VENT PAM	4(F)	(LTR)		3.20
116.	PM73	WASTE DEWATERING TANK	1(G)	.1-$10^{[illegible]}$MREM/HR		3.21
117.	PM74A	SOLID WASTE CASK A	1(G)	10-$10^{[illegible]}$MREM/HR		3.22
118.	PM74B	SOLID WASTE CASK B	1(G)	10-$10^{[illegible]}$MREM/HR		3.23
119.	PM74C	SOLID WASTE CASK C	1(G)	10-$10^{[illegible]}$MREM/HR		3.24
120.	PM74D	SOLID WASTE CASK D	1(G)	10-$10^{[illegible]}$MREM/HR		3.25
121.	PM77	RADWASTE TANK VENT	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.26
122.	PM79	SALTWATER DRAIN TANK	1(G)	$10^{-[illegible]}$-$10^{-[illegible]}$MC/CC		3.27

S.P.D.S. MAIN MENU - GRAPHIC NO. 10 -TIME-
HHMMSS

REACTIVITY CONTROL

O NO. 11 - APRM LEVELS

\- NO. 12 - CONTROL ROD POSITIONS

REACTOR CORE COOLANT / HEAT REMOVAL

O NO. 13 - RX DOME PRESS, WR/FZ LEVELS, REF. LEG DW TEMP.

O NO. 14 - CORE SPRAY FLOW, SUPP POOL TEMPERATURE

REACTOR COOLANT SYSTEM INTEGRITY

O NO. 15 - RX DOME PRESS. DW PRESS, SUPP POOL PRESS

O NO. 16 - DW AVG TEMP, SUPP POOL WATER LEVELS

O NO. 17 - ADS/SRV POSITIONS, WR/FZ LEVELS

CONTAINMENT INTEGRITY / HEAT REMOVAL

O NO. 18 - RX BLDG FLOOD LEVEL, RX BLDG DELTA PRESS.
DW PRESS, SUPP POOL PRESS, DW AVG TEMP.

O NO. 19 - RX DOME PRESS, SUPP POOL LEVEL, SUPP POOL TEMP.

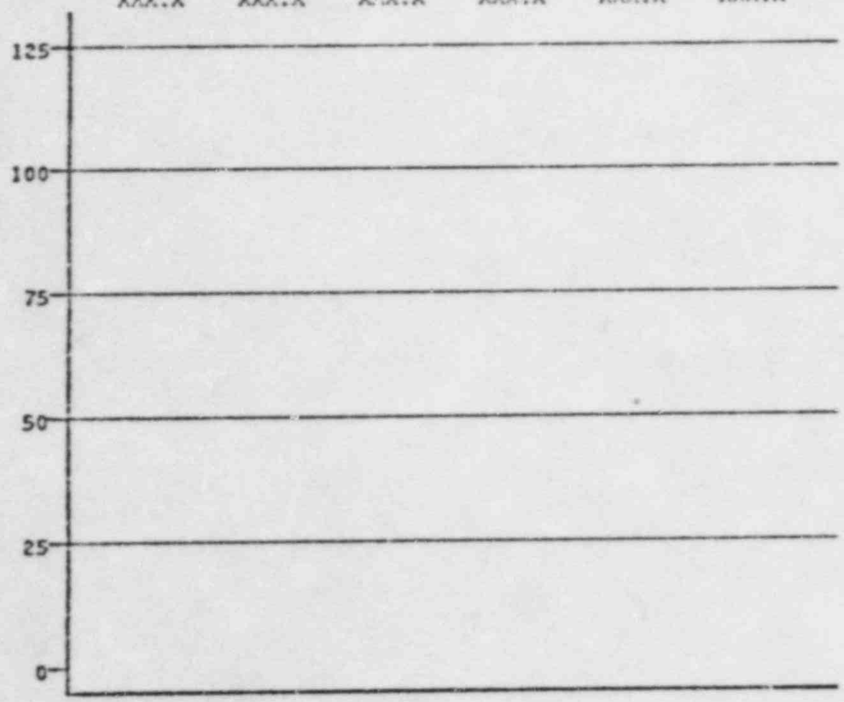
O NO. 20 - DW / SUPP POOL H2 CONCENTRATION

CHART 04/2

REACTIVITY CONTROL - GRAPHIC NO. 11

-TIME-
HHMMSS

	APRM A	APRM B	APRM C	APRM D	APRM E	APRM F
	B000	B001	B002	B003	B004	B005
APRM BYPASSED	A536	A537	A538	A539	A540	A541
	XXX.X	XXX.X	XXX.X	XXX.X	XXX.X	XXX.X



B000 APRM A FLUX LEVEL

B001 APRM B FLUX LEVEL

B002 APRM C FLUX LEVEL

B003 APRM D FLUX LEVEL

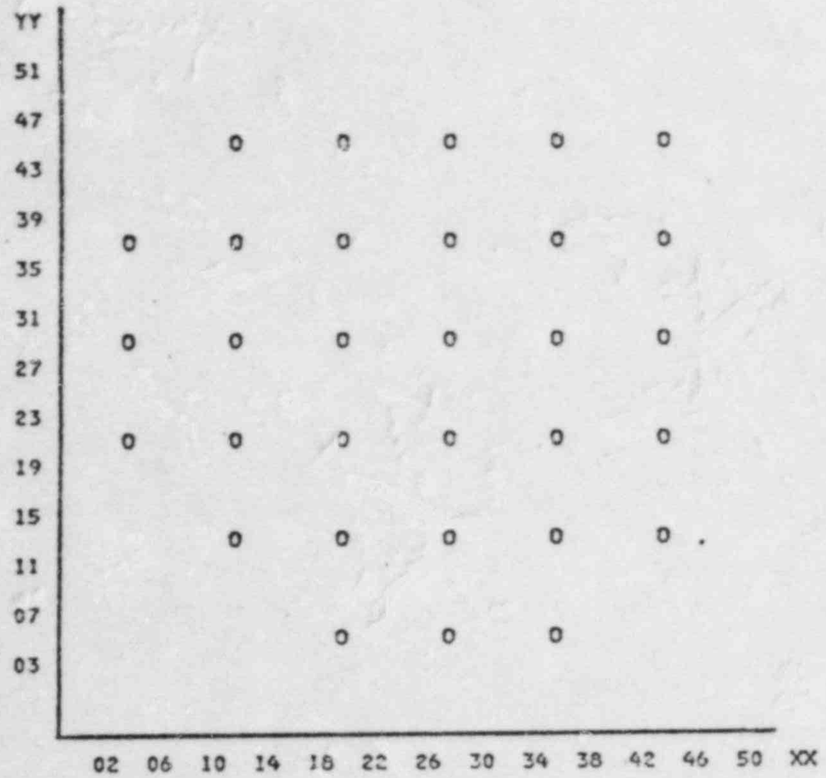
B004 APRM E FLUX LEVEL

B005 APRM F FLUX LEVEL

CHART

04/2

CONTROL ROD POSITIONS - GRAPHIC NO. 12



GRN - FULL IN - 00

YEL - DEEP - 02 THRU 28

CYN - SHALLOW - 30 THRU 46

RED - FULL OUT - 48

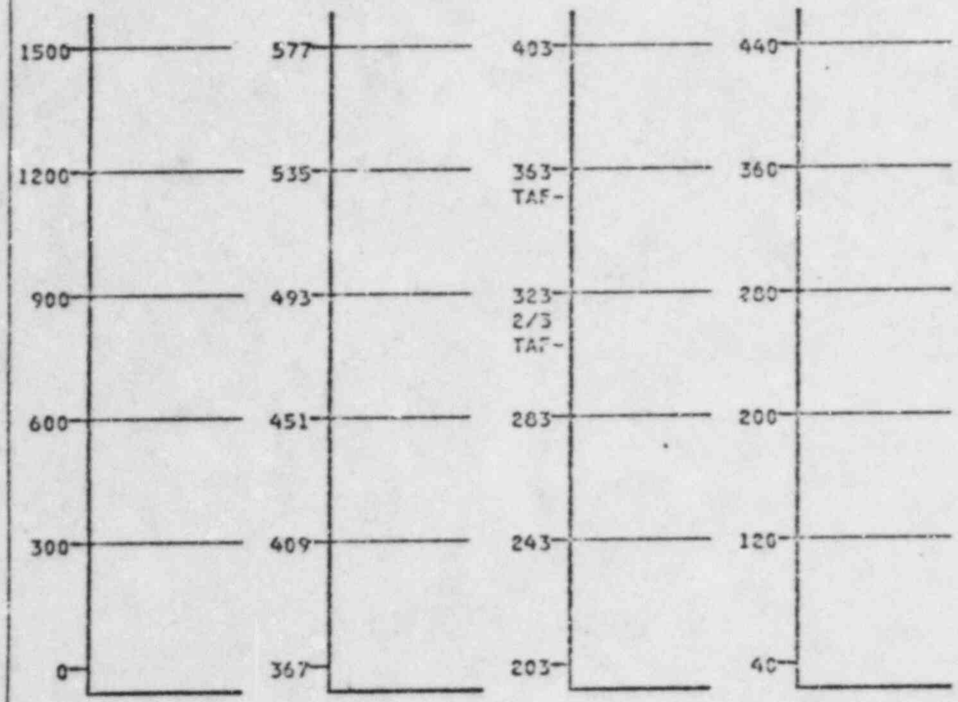
* * - UNKNOWN POSITION

SELECTED ROD WILL BLINK

CHART 04/29

REACTOR CORE COOLANT / HEAT REMOVAL - GRAPHIC NO. 13 -TIME- HH:MM:SS

RX PRESSURE	RX WATER LEVEL (WR)	RX WATER LEVEL (FZ)	DRYWELL TEMPERATURE
A B	A B	A B	A B
L940 L941	L943 L944	E500 E501	L993 L994
XXXX XXXX	XXXX XXXX	XXXX XXXX	XXXX XXXX



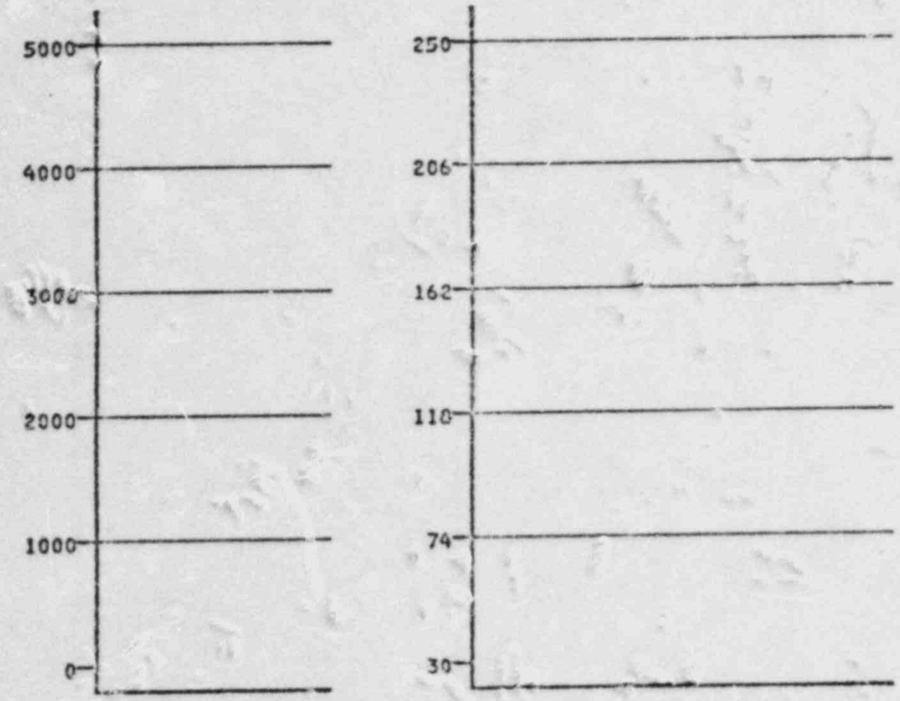
L940 REACTOR PRESSURE A PSIG
L941 REACTOR PRESSURE B PSIG
L943 REACTOR LEVEL A WR INCH
L944 REACTOR LEVEL B WR INCH
E500 REACTOR LEVEL A FZ INCH
E501 REACTOR LEVEL B FZ INCH
L993 DW T EL 145 AZ 55 DEGF
L994 DW T EL 145 AZ 230 DEGF

CHART

04/29/

REACTOR CORE COOLANT / HEAT REMOVAL - GRAPHIC NO. 14 -TIME- HH:MM:SS

CORE SPRAY FLOW		SUPPRESSION POOL WATER TEMP			
A	B	1	2	3	4
G714	G718	L625	L626	L925	L926
XXXXX	XXXXX	XXXX	XXXX	XXXX	XXXX



G714 CORE SPRAY SYSTEM A GPM

G718 CORE SPRAY SYSTEM B GPM

L625 SP QUAD1 2FT WTR T DEGF

L626 SP QUAD2 2FT WTR T DEGF

L925 SP QUAD3 2FT WTR T DEGF

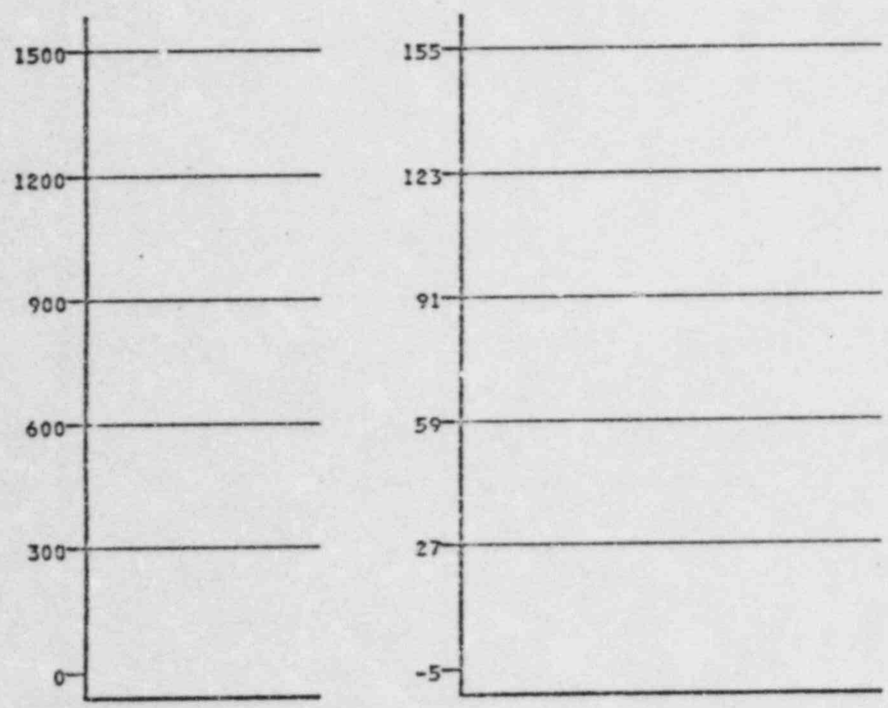
L926 SP QUAD4 2FT WTR T DEGF

CHART

04/

REACTOR COOLANT SYSTEM INTEGRITY - GRAPHIC NO. 15 -TIME- HH:MM:SS

RX PRESSURE		DRYWELL PRESSURE		SP PRESSURE	
A	B	A	B	A	B
L940	L941	F655	F656	F657	F658
XXXX.X	XXXX.X	XXX.X	XXX.X	XXX.X	XXX.X



L940 REACTOR PRESSURE A PSIG

L941 REACTOR PRESSURE B PSIG

F655 DRYWELL WR P A PSIG

F656 DRYWELL WR P B PSIG

F657 SUPR CHMBR WR P A PSIG

F658 SUPR CHMBR WR P B PSIG

CHART

04/29.

REACTOR COOLANT SYSTEM INTEGRITY - GRAPHIC NO. 16

-TIME-
HHMMSS

DW AVG TEMP

X1

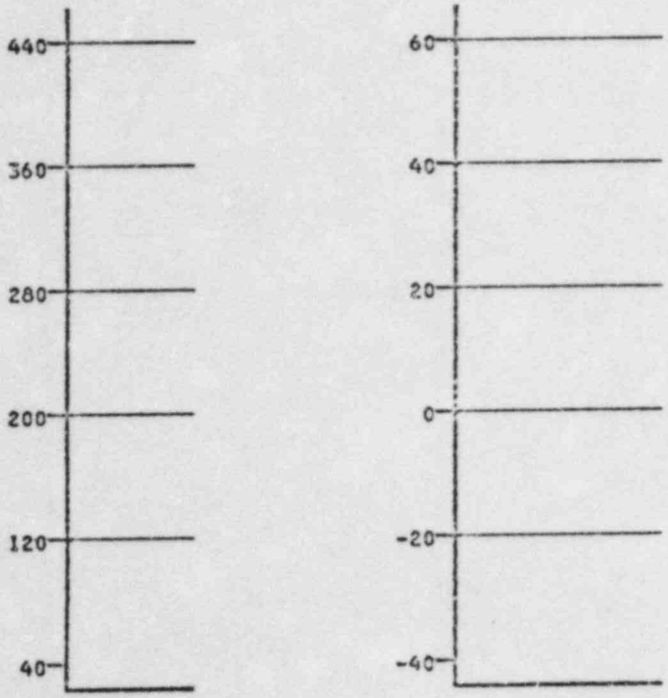
XXX.X

SUPR POOL WATER LEVEL

A B

L927 L928

XXX.X XXX.X



DRYWELL AVG TEMP DEGF

L927 SP WTR LVL A INCH

L928 SP WTR LVL B INCH

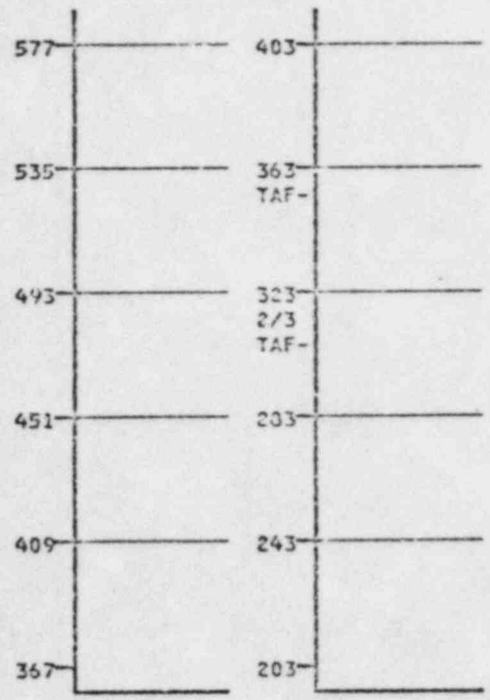
CHART 04/

REACTOR COOLANT SYSTEM INTEGRITY - GRAPHIC NO. 17 -TIME- HH:MM:SS

ADS / SRV TAIL PRESS OPEN / CLOSED

ADS/SRV	POSITION	PSIG
E507 A	XXXX	XXX.X
E511 B	XXXX	XXX.X
E514 C	XXXX	XXX.X
E515 D	XXXX	XXX.X
E520 E	XXXX	XXX.X
E522 F	XXXX	XXX.X
L947 G	XXXX	XXX.X
L948 H	XXXX	XXX.X
E527 J	XXXX	XXX.X
E528 K	XXXX	XXX.X
G703 L	XXXX	XXX.X

RX WATER LEVEL (NR) A L943	RX WATER LEVEL (NR) B L944	RX WATER LEVEL (FZ) A E500	RX WATER LEVEL (FZ) B E501
XXX.X	XXX.X	XXX.X	XXX.X



L943 REACTOR LEVEL A NR INCH

L944 REACTOR LEVEL B NR INCH

E500 REACTOR LEVEL A FZ INCH

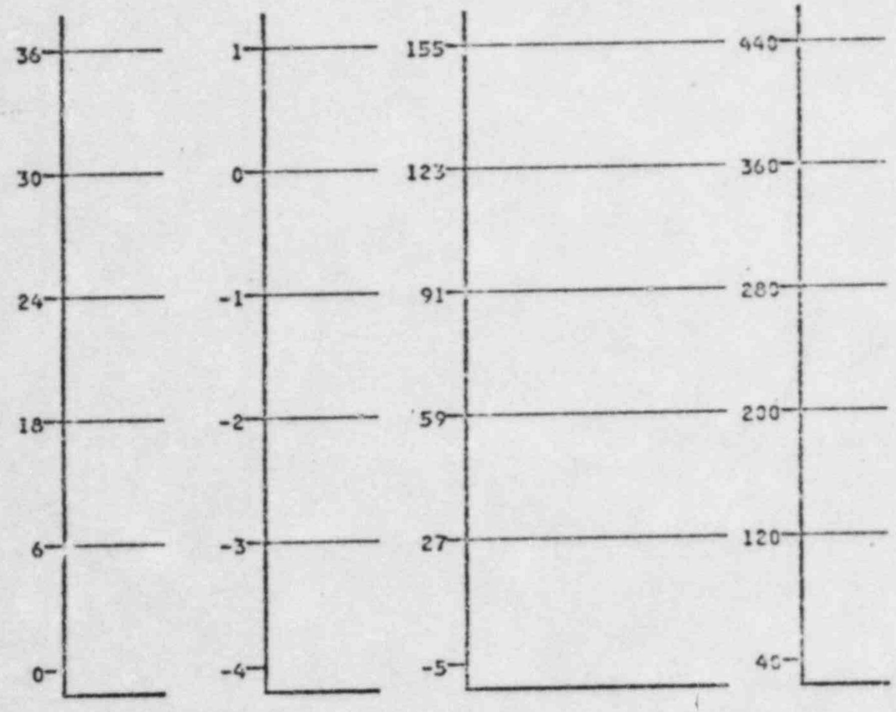
E501 REACTOR LEVEL B FZ INCH

CHART

04/29

CONTAINMENT INTEGRITY / HEAT REMOVAL - GRAPHIC NO. 18 -TIME-
HHMMSS

RB FLOOD LEVEL	RB DELTA PRESS	DW PRESS		SP PRESS		DW AVG TEMP
		A	B	A	B	
F653	G715	F655	F656	F657	F658	XI
XXX.X	XXX.X	XXX.X	XXX.X	XXX.X	XXX.X	XXX.X



F653 RB FLOOD LVL A INCH

G715 REACTOR BLDG P IN H2O

F655 DRYWELL WR P A PSIG

F656 DRYWELL WR P B PSIG

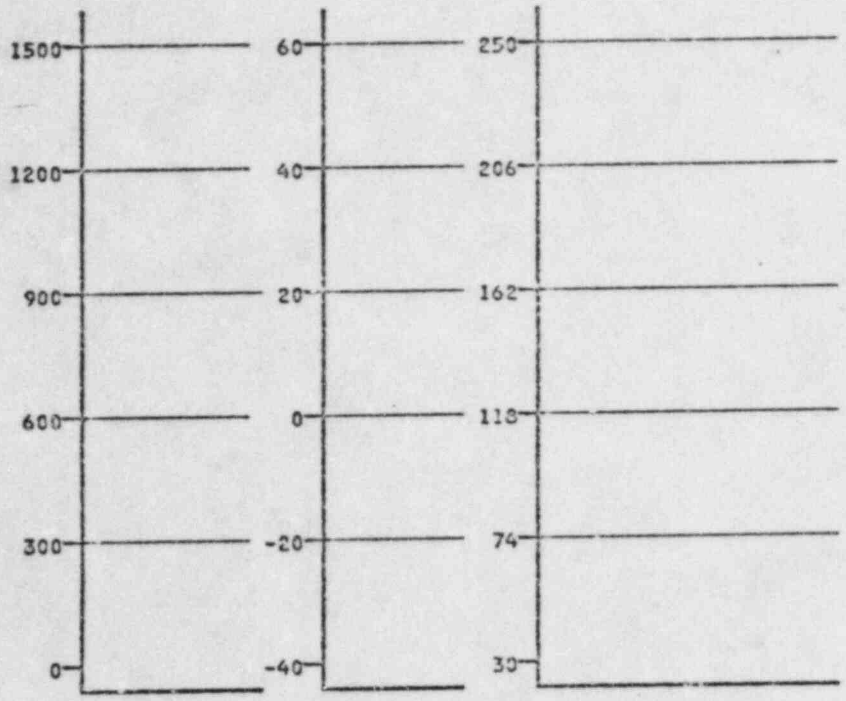
F657 SUPR CHMBR WR P A PSIG

F658 SUPR CHMBR WR P B PSIG

DRYWELL AVG TEMP DEGF

CONTAINMENT INTEGRITY / HEAT REMOVAL - GRAPHIC NO. 19 -TIME-
HHMMSS

REACTOR PRESSURE		SP WATER LEVEL		SP WATER TEMPERATURE			
A	B	A	B	1	2	3	4
L940	L941	L927	L928	L625	L626	L925	L926
XXXX	XXXX	XXXX	XXXX	XXXX	XXXX	XXXX	XXXX



L940 REACTOR PRESSURE A PSIG

L941 REACTOR PRESSURE B PSIG

L927 SP WTR LVL A INCH

L928 SP WTR LVL B INCH

L625 SP QUAD1 2FT WTR T DEGF

L626 SP QUAD2 2FT WTR T DEGF

L925 SP QUAD3 2FT WTR T DEGF

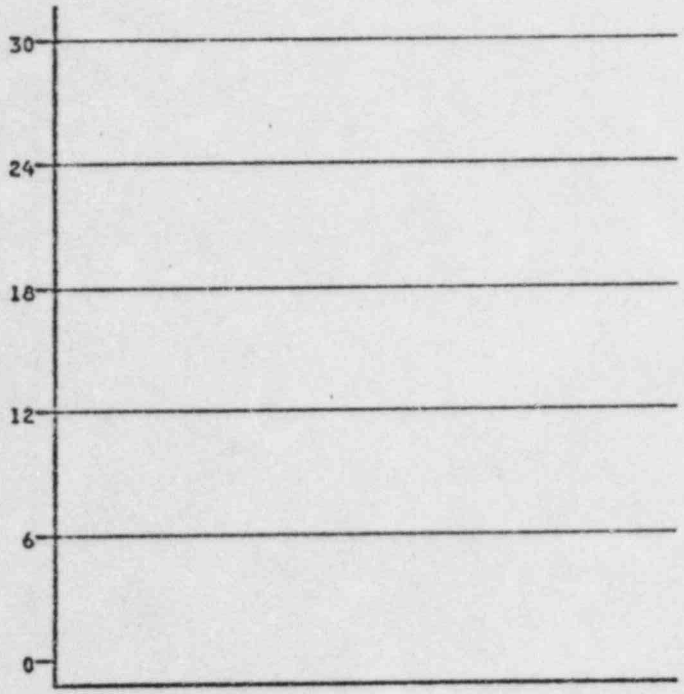
L926 SP QUAD4 2FT WTR T DEGF

CHART

04/2

CONTAINMENT INTEGRITY / HEAT REMOVAL - GRAPHIC NO. 20 -TIME-
HH:MM:SS

DW H2 CONCENTRATION		SP H2 CONCENTRATION	
A	B	A	B
L929	L938	L939	E502
XX.XX	XX.XX	XX.XY	XX.XX



L929 DRYWELL H2 CONC A %   L939 SUPR CHMBR H2 CONC A %

L938 DRYWELL H2 CONC B %   E502 SUPR CHMBR H2 CONC B %

APPENDIX H
ERF Phase I Special Log, Format and Frequency

LOG 1 — REACTOR PARAMETERS

TIME	WIDE RANGE RX LEVEL A INCH L943	WIDE RANGE RX LEVEL B INCH L944	FUEL ZONE RX LEVEL A INCH E500	FUEL ZONE RX LEVEL B INCH E501	NEUTRON FLUX LEVELS APRM A % PWR B000	APRM B % PWR B001	APRM C % PWR B002	APRM D % PWR B003	APRM E % PWR B004	APRM F % PWR B005	# RODS .NOT. FULL-IN NUMBER C499
114429			-352.0	-357.3	-0.2	-0.4	-0.2	-0.7	-0.3	-0.2	4
← SCAN FREQUENCY	1s	15s	1s	15s	1s	1m	1s	1m	1m	1m	

m= minute
S= second

LOG 2 — AUTOMATIC DEPRESSURIZATION SYSTEM

TIME	REACTOR VESSEL WR PRESS A PSIG L940	WR PRESS B PSIG L941	SAFETY RELIEF VALVE TAILPIPE PRESSURES SRV A PSIG E507	SRV B PSIG E511	SRV C PSIG E514	SRV D PSIG E515	SRV E PSIG E520	SRV F PSIG E522	SRV G PSIG L947	SRV H PSIG L949	SRV J PSIG E527	SRV K PSIG E528	SRV L PSIG G703	SRV N2 HYD A PRESS PSIG L974	N2 HYD B PRESS PSIG L977
114429	-29.9	-27.9	89.9				-25.2	-25.6	-25.7			-25.6		-30.1	
← SCAN FREQUENCY	1s	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m

LOG 3 — EMERGENCY CORE COOLING SYSTEM PARAMETERS

TIME	COOLANT FLOWS RCIC PP DISCH GPM G704	HPCI PP DISCH GPM G713	RHR SYS A FLOW GPM G708	RHR SYS B FLOW GPM G709	CORE SPRAY A GPM G714	CORE SPRAY B GPM G718	COOLANT TEMPERATURES RHR HX A OUT DEG.F G701	RHR HX A IN DEG.F G705	RHR HX B OUT DEG.F G702	RHR HX B IN DEG.F G706	RHR HX A FLOW GPM G719	RHR HX B FLOW GPM F652	RHR HX A OUT TEMP DEG.F G710	RHR HX B OUT TEMP DEG.F G711
114429	44.0	-660.0	-1015.5	883.9		-938.3	24.4	70.1	77.6	74.9	9806.4	3495.3	61.1	58.1
← SCAN FREQUENCY	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m

LOG 4 — CONTAINMENT PARAMETERS - DRYWELL

TIME	PRESSURE WIDE RANGE A PSIG F655	WIDE RANGE B PSIG F656	TEMPERATURES ELEV 68 FT AZ 160 DEG.F L777	ELEV 68 FT AZ 320 DEG.F F602	ELEV 80 FT AZ 000 DEG.F L825	ELEV 83 FT AZ 25 DEG.F L911	ELEV 83 FT AZ 145 DEG.F L956	ELEV 83 FT AZ 265 DEG.F L957	ELEV 102 FT AZ 190 DEG.F L996	ELEV 102 FT AZ 350 DEG.F L997	ELEV 132 FT AZ 55 DEG.F L993	ELEV 132 FT AZ 230 DEG.F L994	ELEV 162 FT TDC RPV DEG.F G562
114429	-0.3	-2.3	80.4	76.4	79.7	78.1	83.7	78.5	91.5	76.9	79.7	77.9	75.8
← SCAN FREQUENCY	1s	1s	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m	1m

10-15-82

LOG 5 — CONTAINMENT PARAMETERS - SUPPRESSION CHAMBERS

TIME	PRESSURE WIDE RANGE A PSIG F657	WIDE RANGE B PSIG F658	WATER LEVEL A INCH L927	WATER LEVEL B INCH L928	WATER TEMPERATURES QUAD 1 1 FOOT DEPTH DEG.F F659	QUAD 2 1 FOOT DEPTH DEG.F L622	QUAD 3 1 FOOT DEPTH DEG.F L623	QUAD 4 1 FOOT DEPTH DEG.F L624	QUAD 1 2 FOOT DEPTH DEG.F L625	QUAD 2 2 FOOT DEPTH DEG.F L626	QUAD 3 2 FOOT DEPTH DEG.F L925	QUAD 4 2 FOOT DEPTH DEG.F L926	
114429	-0.6	-0.6	-28.3	-51.8		66.3		69.3			213.7		
	1s	1s	15s	1m	1m	1m	1m	1m	1m	1m	1m	1m	← SCAN FREQ.

LOG 6 — CONTAINMENT PARAMETERS

TIME	DRYWELL H2 CONC A % L929	H2 CONC B % L938	O2 CONC A % E503	O2 CONC B % E504	SUPPRESSION CHAMBER H2 CONC A % L939	H2 CONC B % E502	O2 CONC A % E505	O2 CONC B % L973	REACTOR BUILDING FLOOD LEVEL A INCH F653	FLOOD LEVEL B INCH F654	PRESS IN H2O G715	
114429	-2509.8	-2525.5	-2529.7		-2566.3	-2465.2	-2485.1	-2522.7	-9.4	-9.5	0.3	
	15s	1m	15s	1m	15s	1m	15s	1m	1m	1m	15s	← SCAN FREQ.

LOG 7 — BALANCE OF PLANT SYSTEM PARAMETERS

TIME	RBCLCW SUPPLY HDR A TEMP DEG.F L851	SUPPLY HDR B TEMP DEG.F L852	FEEDWATER CND STOR TK LEVEL FEET F593	LOOP A FLOW MLB/HR B022	LOOP B FLOW MLB/HR B023	LOOP A TEMP 1 DEG.F B050	LOOP A TEMP 2 DEG.F B051	LOOP B TEMP 1 DEG.F B052	LOOP B TEMP 2 DEG.F B053	CONDENSER E-15A PRESS IN HG VAC F507	E-15B PRESS IN HG VAC F508	CIRCULATING WATER PUMP A DISCH PRESS PSIG L813	PUMP B DISCH PRESS PSIG L814	PUMP C DISCH PRESS PSIG L815	PUMP D DISCH PRESS PSIG L816	
114429	65.4	71.2	41.6	0.9	-1.9					0.9	0.5	0.5	0.4	-0.7		
	1m	1m	1m	15s	15s	15s	1m	15s	1m	1m	1m	1m	1m	1m	1m	← SCAN FREQ.