John D. O'Toole
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003
Telephone (212) 460-2533

November 4, 1983
Re: Indian Point Unit No. 2
Docket No. 50-247

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

Dear Mr. Eisenhut:

This letter is in response to your letter dated July 8, 1983 and your clarification letter dated October 21, 1983 concerning "Required Actions Based on Generic Implications of Salem ATWS Events" (Generic Letter 83-28). Your letters requested a response by November 5, 1983 providing the status of each licensee's current conformance with the positions contained therein, and plans and schedules for any needed improvements for conformance with the positions.

Attachment 1 to this letter provides a description of our programs and plans with respect to the positions identified by your staff. Based on the results of our current review, the actions taken immediately following the Salem event, as indicated in our response to IE Bulletin No. 83-01 and the history of reliable reactor trip breaker operation at Indian Point Unit No. 2, we conclude that Indian Point Unit No. 2 can be operated without undue risk to the health and safety of the public.

Based on the results of our review, additional effort will be required to fully assess the extent to which we comply and, as necessary, to develop plans and schedules for resolution of non-conformances.

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We anticipate completion of our detailed review during February 1984 and expect to provide NRC a complete status of our plans and schedule in response to this issue at that time.

Very truly yours,

John D. O'Toole
Vice President

cc: Mr Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Mr. Thomas Foley, Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, New York 10511

Subscribed and sworn to
before me this 4th day
of November, 1983.

Notary Public

CONRAD TROMBA
Notary Public State of New York
No. 30-4022875
Qualified in Nassau County
Terms expires March 30, 1985

Attachment 1

Response to Generic Letter 83-28
"Required Actions Based on Generic
Implications of Salem ATWS Events"

Consolidated Edison Company of New York Inc
Indian Point Unit No. 2
Docket No. 50-247

November, 1983

1.1 Post Trip Review (Program Description and Procedure)

Position

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

Item 1. The criteria for determining the acceptability of restart.

Response to Item 1.

The criteria for determining the acceptability of restart are identified in Plant Startup Check Off List 46 (COL-46). The specific criteria dealing with restart are contained in Section E, "Administrative" which require that in the event the shutdown was a result of a unit trip, the cause of the trip must be determined prior to restart. The procedure requires that the first out annunciator is recorded, the station computer sequence of events log is obtained and attached to the COL, and the first out annunciator and sequence of events cause of trip are reconciled. The plant may be restarted if the SWS and STA agree as to the cause of the trip, otherwise plant restart requires SNSC approval. If the cause of the trip cannot be reconciled, the SWS and STA are directed to Station Administrative Order (SAO) 131, "Station Nuclear Safety Committee." SAO 131 Section 4.1.1 states that Station Nuclear Safety Committee (SNSC) approval must be obtained prior to bringing the plant critical if the three point startup criteria are not met. The three point startyp criteria require that both the Senior Watch Supervisor and the Shift Technical Advisor have idependtly 1) positively identified the cause of the reactor trip, 2) determined that startup does not involve unusual conditions and 3) believe that a potential safety hazard exists. This policy was established piror to the Salem event.

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

Response to Item 2

Responsibility/authority for performing the initial post-trip review is delegated to the SWS and STA in accordance with COL-46. The SWS reports through the normal plant chain of command to the Chief Operations Engineer and General Manager of Nuclear Power Generation while the STA serves as an advisor to the SWS and reports independently to the General Manager, Technical Support. The plant may be restarted if the SWS and STA agree as to the cause of the trip, otherwise plant restart requires SNSC approval.

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

Response to Item 3.

Personnel performing the initial post-trip reviews, must be qualified as SWSs and STAs. SWSs hold Senior Operator Licenses and training requirements for the SWS include those required for qualification as a Senior Reactor Operator (Senior Operator license holder) plus additional training emphasizing the SWS's responsibility for safe operation and the decision making function he is to provide for assuring safety. The STA is a degreed engineer who has no responsibility for plant operation and is independent of the plant operating organizations. STA training requirements are in accordance with NUREG-0737 item I.A.1.1 requirements as described in a letter to NRC (Darrell G. Eisenhut) from Con Edison (John D. O'Toole) dated February 26, 1981.

STAs and SWSs received, as part of their requalification program, training in post trip review. Licensed operator qualification and requalification training programs will be modified to require training in post trip review.

The Station Nuclear Safety Committee functions to advise the Vice President, Nuclear Power on all matters related to nuclear safety. Its composition and qualifications are more than equivalent to the Unit Review Group specified in the Standard Technical Specifications.

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

Response to Item 4.

The principal sources of plant information used to conduct the initial review include the sequence of events log printout, the record identifying the first out annunciator alarms received and operator observations. In accordance with COL-46, if the shutdown was the result of a trip, the cause of the trip must be determined prior to start-up, the first out annunciator alarms recorded and the sequence of events log obtained and attached to COL-46. When applicable, other sources, listed in response to item 1.2, are used to support the post-trip review.

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

Response to Item 5.

The methods for comparing the event information with known or expected plant behavior begins with the initial abnormal and/or emergency operating procedural requirements of verifying that all required automatic actions have occurred.

Prior to restart, the sequence of events log printout is reviewed by both the SWS and STA to determine/verify the cause of the trip. In addition to reconciling the first out annunciator cause of trip with the sequence of events printout cause of trip, each of the alarms identified on the printout is reviewed to determine if the alarm was received as/when expected. Alarms received that were not expected or that were not received but were expected are reviewed to determine the cause of the actuation. Methods for performing these initial reviews are currently a matter of experience for the SWS and STA as well as other on-site plant staff and will be incorporated in appropriate procedures. When a question exists as to a particular aspect of the plant response to the trip, other plant personnel or the off-site engineering staff consulted. If necessary, the NSSS supplier, A/E or other outside organizations, are requested to review the circumstances of the event. Reviews may include a comparison of the event to FSAR analyses, more recent generic analyses, or as necessary, preparation of revised plant specific analyses. Such reviews are requested and performed on a case by case basis, as identified and as needed, and are not currently required by administrative procedures.

We are participating in the development of the INPO good practice OP 211 "Post-Trip Reviews." A revised post trip review procedure will be developed incorporating the recommendations of the final good practice.

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

Response to Item 6.

The criteria for determining the need for independent assessment of an event are contained is SAO-131. The plant may be restarted if the SWS and STA agree as to the cause of the trip, otherwise plant restart requires SNSC approval.

Preservation of evidence to support independent assessment of an event is supported by the requirement to attach the sequence of events printout to COL-46. Record keeping requirements contained in "Nuclear Power Station Quality Assurance Record Management Program," SAO-121, provide for retention of operating records.

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

Response to Item 7.

Copies of the referenced documents are contained in Appendix A to this submittal.

1.2 Post Trip Review (Data and Information Capability)

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.

A report shall be prepared which describes and justifies the adequacy of equipment for diagnosing an unscheduled reactor shutdown. The report shall describe as a minimum:

Item 1. Capability for Assessing Sequence of Events

Response to Item 1.

The following describes the plant's computer which provides the sequence of events reports and pre/post trip logs. The computer is a Westinghouse model Proteus 2500 Computer System. The computer system configuration consists of five computers interconnected through a shared common memory subsystem. The system includes keyboards, CRTs and Operator consoles with a control module. A sequence of events (SOE) function monitors Reactor Protection System contacts. Table 1.2-1 provides a listing of all the variables that are presently monitored. The system presently monitors 56 SOE inputs and provides a time recording resolution capability of one millisecond. If more than one status change occurs within one millisecond, the events are still detected and recorded. The SOE report format maximum limit is 64 events or 66 seconds of events, whichever comes first. If the buffer collection SOE inputs become full, the computer reads the data immediately so that SOE additional information is retained. A sample sequence of events report is attached as Table 1.2-2. The capability to store data on magnetic tapes is currently being implemented. When fully functional, the capability will allow storage of data for an indefinite period of time. The power supply for the plant computer is non-Class 1E; however, it is powered by an uninterruptible power supply with battery and diesel generator backup. At present, the diesel generator is being prepared for service.

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

Response to Item 2.

The major plant capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns and the functioning of safety related equipment are:

- Plant Process Computer
- Plant Recorders (strip charts)

Plant Process Computer (Westinghouse Model Proteus 2500)

The plant computer also contains a program for printing out the value of preselected groups of variables collected over preselected time intervals immediately preceding and following a reactor trip. Table 1.2-3 provides a listing of the analog inputs to the trip logs. The trip logs provide a time history of various analog variables preceding and following a reactor trip.

The Pre-Trip logs collect data continuously in a circular buffer until the trip signal is received at which time the data in the buffer is printed out on the Computer Room Line Printer.

The Post-Trip logs are inactive until a trip signal is received, at which point they begin to collect data. This data is then printed on the Computer Room Line Printer. The capability to store data on magnetic tapes is currently being implemented. When fully functional, this capability will allow storage of data for an indefinite period of time.

The trip log records 40 data entries prior to the reactor trip and 40 data entries after the trip for each variable during the sample period. The duration of time history (minutes before the trip and after the trip) and sampling rates are noted on Table 1.2-3.

An example of printout display is provided in Table 1.2-4. A listing of parameters monitored and format of printout display is provided in Tables 1.2-3 and 1.2-4.

The basis for selecting parameters was engineering evaluation as to which parameters or trending information was important to assess safety functions and equipment performance. The sampling rate was based upon equipment capability, consideration of tradeoffs between sample period and sample rate, and engineering evaluation of the speed at which variables may change.

Plant Recorders

A list of recorders installed in the Control Room and the parameters monitored are identified in Table 1.2-5. These recorders are continuous monitoring devices and their response characteristics are typical of instruments in this application. They provide the operator ability to identify trend changes during a particular transient and provide significant information regarding the course of events, equipment performance prior to, during and after a transient.

The basis for selecting parameters was engineering evaluation as to which parameters or trending information was important to assess safety functions and equipment performance. Recorder speed was based upon standard characteristics for instruments of this type.

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

Response to Item 3

Various sources of information are available to the operator for assessment of plant variables and post trip review. This information is sufficient to diagnose the cause(s) of unscheduled reactor shutdowns prior to restart and to ascertain the proper functioning of safety related equipment. The following lists the major displays presently employed for this analysis:

- First Out Annunciator Panel Indications
- Protective Relay Targets
- Operator's Observations prior to and during the event
- Plant Computer Report
- Plant Recorders
- Accident Assessment Panels
- Rudimentary Safety Parameter Display System

This data is supplemented, when necessary by the following:

- Inspection of plant equipment to determine the root cause of failure.

- Inspection of local annunciator panels to determine the cause of equipment malfunctions.
- Special testing to detect equipment malfunctioning or improperly operation.

As discussed previously, recording the first out annunciator indication following a reactor trip is required by procedure. A list of alarm annunciators provided on the reactor and turbine first out annunciator panels is provided in Table 1.2-6. Additionally, Plant Startup Check Off List, COL-46, requires completion of the trip portion of the "Electrical Relay Positions Following a Trip and Prior to Startup," Check Off List, COL-1A. The list of relays checked prior to restart is indicated in the attached Check Off List, COL-1A.

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

Response to Item 4

No changes are planned.

2.1 Equipment Classification and Vendor Interface (Reactor Trip System Components)

Position

Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement. In addition, for these components, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of these components should be contacted and an interface established. Where vendors can not be iden-

tified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and nonnuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.

Response to Position

Based on a review of the systems and procedures in use at Indian Point Unit 2, as discussed in the response to Position 2.2, it is concluded that the reactor trip system components are identified as Class-A and would be identified as such on work control documents.

See Section 2.2 for additional details.

2.2 Equipment Classification and Vendor Interface (Programs for all Safety-Related Components)

Position

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

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

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

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

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

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

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

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

Response to Item 1

Those systems which contain safety-related components are designated as Class-A. Class-A Systems are listed in Exhibit A to Con Edison's Corporate Instruction-240-1 entitled "Quality Assurance Program for Operating Nuclear Plants."

Systems currently classified as Class A are identified in a letter dated March 11, 1983 from Con Edison (John D. O'Toole) to NRC (Steven A. Varga). Specifically, the determination of which systems, structures and components affect safety....shall include those:

- which comprise or are necessary to insure the integrity of the reactor coolant pressure boundary

- which ensure the capability to shutdown the reactor and maintain it in a safe shutdown condition

- whose failure could result in conservatively calculated offsite doses that exceed 0.5 Rem to the whole body or the equivalent to any part of the body

- structures whose failure could reduce the functioning of plant features within the above categories to an unacceptable safety level

Within Con Edison's Quality Assurance Program, provision is made to declassify portions of a system or components within a system. The off-site engineering staff is required to evaluate requests for declassification using the criteria identified above.

Upon completing such an evaluation and concluding that a component may be declassified, a memorandum is issued for concurrence by Nuclear Power.

Upon concurrence a copy of the memorandum is placed on file with Nuclear Power and the Central Files Controller at the plant site and is sent to a standard,predetermined distribution as well as job specific personnel.

Class A maintenance and modification activities are controlled in accordance with Station Administrative Order SAO-104 "Maintenance Work Request (MWR) Procedure." In accordance with SAO-104, an MWR can only be authorized for implementation, by the General Manager, Nuclear Power Generation, the Chief Operations Engineer, Maintenance Engineer or Instrument and Control Engineer. By procedure (SAO-104), the authorizer is required to classify the MWR. Personnel required to identify work as safety-related at the plant site are directed by procedure to make that determination by reviewing the list of safety-related (Class A) structures, systems and components contained in Exhibit A to Corporate Instruction 240-1 "Quality Assurance Program" for Indian Point Unit No. 2 and the declassification memoranda on file at the plant site. Upon making this determination, the MWR authorizer is required to indicate the classification in the space provided on the MWR form.

Once an MWR or modification has been classified Class A, all activities associated with that MWR or modification will be controlled in accordance with Con Edison's QA program for Indian Point Unit No. 2. This includes all those activities defined in the introduction to 10 CFR 50 Appendix B including maintenance, surveillance and parts replacement. In addition, modification work initiated by Engineering must be classified before the modification can be implemented. This is required by Engineering Procedure OP-290-1 and is accomplished in a manner similar to that described above for MWRs.

Preventive maintenance of safety-related instrumentation and controls and surveillance tests required by Technical Specifications are accomplished in accordance with written procedures which require approval by the Station Nuclear Safety Committee in accordance with SAO-102 entitled "Procedure/Procedure Change Approval Policy." All procedures/procedure changes subject to SNSC approval are required to identify such SNSC approval on their cover page. Such designation is considered to serve to identify the procedure/procedure change as a safety-related document.

Con Edison's Quality Assurance program for Indian Point Unit No. 2 describes the management controls relied upon to assure that administrative program provisions are adhered to. These include monitoring of these programatic requirements by Quality Assurance personnel as well as periodic audits performed by Quality Assurance Auditors under the auspices of the Nuclear Facilities Safety Committee (plant specific company Nuclear Review and Audit Group (CNRAG)). These activities include inspection and review of maintenance, procurement and modification procedure processes.

All procurement requests for components which have been classified as Class-A are reviewed and, as required by the design bases, 10 CFR 50 or other NRC requirements, design verification and qualification testing is specified. When qualification testing is required, the applicable specifications identifying expected safety service conditions and the qualification documents identifying the limits of life identified by the manufacturer are requested. This is accomplished as equipment qualification requirements are identified and new equipment is procured.

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

Response to Item 2

Con Edison and its NSSS vendor have increased the controls in their safety information interface program.

Station Administrative Order 120, "Nuclear Plant Operating and Safety Information Handling System," assigns responsibility for receipt, dissemination and tracking of safety related information to the Director, Regulatory Affairs, Nuclear Power. The NSSS vendor uses a receipt of acknowledgement system to ensure safety-related information is received by the Director of Regulatory Affairs. Further, the NSSS vendor is issuing on a periodic basis indices of Bulletins and Letters to enable Con Edison to assure identification of each Bulletin and Letter that has been issued. The NSSS Vendor also reissued a complete set of these documents. All those related to the reactor trip breakers were received and incorporated into appropriate procedures.

When any safety-related information is received by Regulatory Affairs it is screened for action level and schedule, addressees are identified, and the package is logged, disseminated and tracked. After initial action is complete and returned to Regulatory Affairs, the item may be closed, referred for safety committee review and/or additional action. The procedure provides for following the status of each item untill closed.

Consolidated Edison is currently participating in the INPO sponsored Nuclear Utility Task Action Committee (NUTAC) working to develop a generic industry response to this issue. Upon its completion, Con Edison will review the final report of that committee, as well as any NRC comments concerning the positions contained therein, for applicability to Indian Point Unit No. 2, and will advise NRC of the planned actions following that review.

3.1/3.2 Post-Maintenance Testing (Reactor Trip System Components (All Other Safety- Related Components))

Position

The following actions are applicable to post maintenance testing:

Item 1

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

Response to Item 1

Post-maintenance testing is performed in accordance with Station Administrative Order, SAO-104, Maintenance Work Request (MWR) Procedure. That procedure requires that the Test and Performance Engineer review the MWR and scope of work, determine the testing requirements (including acceptance criteria) and prepare a post-maintenance test where required. For work performed by the Instrument and Control Section, the I&C Engineer determines the need for post-maintenance testing prior to return to service.

The acceptance criteria on post-maintenance tests are delineated by the Test and Performance Engineer. He uses existing Technical Specification criteria, Surveillance Tests, ASME Section XI requirements, other code and regulatory requirements and essential acceptance criteria delineated in Modification Procedures to demonstrate operability and/or the capability of performing the intended safety functions. The Test and Performance Engineer may also consult Engineering for additional acceptance criteria. The acceptance criteria for tests performed for I&C work are developed by the I&C Engineer in a similar manner.

Technical Specifications and Station Operating Procedures require that specific systems and/or equipment be determined operable prior to bringing the reactor above specified operating conditions. Inherent in the definition of operability is the requirement that the equipment be demonstrated capable of performing its intended safety function. Successful completion of the appropriate Technical Specification

Surveillance Test is considered to constitute such demonstration. The Technical Specification requirement that equipment be demonstrated operable is implemented following maintenance activities through an evaluation of the need for post-maintenance testing by either the Test and Performance or I&C Engineer. Where the scope of the work involved is deemed to require a demonstration that the equipment is capable of performing its safety function and a Technical Specification Surveillance Test exists, the Surveillance Test or an applicable portion thereof is performed. ASME Section XI requirements, other code and regulatory requirements and essential acceptance criteria delineated in Modification Procedures are employed where no Technical Specification Surveillance Test exists to cover the work performed.

A generic system of post-maintenance tests is being developed to insure procedures are reviewed in advance and available and that those procedures are uniformly implemented with respect to the type of work involved.

In addition the post-maintenance testing program outlined in SAO-104 is undergoing extensive review and revision.

Accordingly, post-maintenance testing of the Reactor Trip System as well as post-maintenance testing of all other safety-related equipment is required to be accomplished in accordance with the provisions described above.

Item 2 Licensees and applicants shall check 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.

Response to Item 2

This issue, and Item 1 above, are closely coupled with items 2.1 and 2.2 regarding vendor interface; therefore, with the exception of the recommendations implemented for the reactor trip breakers as described in response to 4.1. Con Edison will respond to this issue following its review of the results of INPO's Nuclear Utility Task Action Committee (NUTAC).

In addition, with respect to the reactor trip (DB-50) breakers, Con Edison has in conjunction with the Westinghouse Owner's Group contracted with Westinghouse for the compilation of all existing maintenance information regarding Westinghouse reactor trip breakers, including lessons learned from the Salem events. Con Edison will review the results of that effort for incorporation into test and maintenance procedures as appropriate, upon receipt of that information.

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

Response to Item 3

At this time Con Edison has not identified post-maintenance testing requirements in existing Technical Specifications which we believe could be demonstrated to degrade rather than enhance safety.

4.1 Reactor Trip System Reliability (Vendor Related Modifications)

Position

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

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

Response to Position 4.1

Con Edison has implemented Westinghouse recommended breaker modifications described in Westinghouse Letter IPP 83-554 and Modification Notice NCD-Elec-18. The recommendations of Westinghouse Technical Bulletin 83-02 have been incorporated except for the number of times the final operational check of the UV attachment is performed. We have concluded that, based on our experience, repearing the check ten times is excessive and we will continue to perform this check three times. The test referred to in Technical Bulletin 83-03 is performed during a refueling outage. This test will be modified to perform independent verification of the undervoltage and shunt trip mechanisms.

Westinghouse Technical Bulletin 83-02

This bulletin recommends a specific lubrication and cleaning procedure which is believed to increase the reliability of the RTBs. This bulletin also recommends a higher frequency of testing and a more stringent test of the UV coils.

Westinghouse Letter IPP 83-554

This Westinghouse letter recommended the removal of the unused overcurrent trip bar brackets.

Westinghouse Modification Notice NCD-Elec-18

This modification called for the replacement of the UVTA in all pre 1972 Westinghouse DB-50 breakers. This modification was implemented by Westinghouse during the construction of Indian Point No. 2 and was reverified (per Westinghouse instruction) during a recent Unit shutdown.

4.2 Reactor Trip System Reliability (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers)

Position

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

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

Response to Item 1

The periodic maintenance program for the reactor trip and bypass breakers currently in effect at Indian Point Unit No. 2 was described in our March 7, 1983 letter responding to IE Bulletin No. 83-01 "Failure of Reactor Trip Breakers (Westinghouse DB-50) To Open On Automatic Trip Signal." The program is contained in a procedure entitled "Reactor Trip and Bypass Breakers Air Circuit Breaker DB-50 P/M Semi Annual Inspection" (MP-16.33).

The program reflects the vendor recommendations in the areas of maintenance frequency and types of lubrication applied. Con Edison is currently evaluating maintenance related recommendations contained in INPO's Significant Operating Experience Report 83-8 relating to the Salem event.

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

Response to Item 2

A program exists to trend the reactor trip breaker performance. The opening time and under voltage dropout voltage are recorded and trended.

The trending information dating back to May, 1980 on the breaker opening time has indicated that there is no substantial change in the breaker opening time.

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

Response to Item 3

Life cycle testing of the shunt trip attachment and the undervoltage trip attachment of the reactor trip breaker is being conducted by Westinghouse for the Westinghouse Owner's Group. This program is aimed toward establishing the service life of these devices, and substantiating periodic test requirements with proper maintenance, replacement and qualification programs. Westinghouse expects the test program to be completed in the second quarter of 1984.

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

Response to Item 4

Con Edison will review the recommendations resulting from the life cycle testing described in response to item 3 above.

4.3 Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)

Position

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

Response to Position

Con Edison will modify the reactor trip system by providing automatic actuation of the reactor trip breaker's shunt trip attachment.

This modification will be installed at the next refueling outage, currently expected to commence during the second quarter of 1984. Con Edison will submit its plant specific design package to NRC for pre-implementation review during the first quarter of 1984 (including any Technical Specification changes required.) That submittal will provide the plant specific information required by NRC's August 10, 1983 SER for Westinghouse plants incorporating the generic automatic shunt trip modification.

4.4 Reactor trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants)

This item is not applicable to Indian Point Unit No. 2.

4.5 Reactor Trip System Reliability (System Functional Testing)

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

Response to Item 1

Con Edison currently performs on-line functional testing of the reactor trip breakers including independent actuation of the shunt trip and undervoltage trip attachments. Actuation of the shunt trip attachment is accomplished through the breaker control switches. Actuation of the undervoltage trip attachment is accomplished in conjunction with testing of the reactor trip logic matrices. In addition, as noted in response to Action 4.3, Con Edison plans to modify the reactor trip system by providing automatic trip system actuation of the shunt trip attachments for the reactor trip breakers. The design of that modification will be consistent with the generic Westinghouse design. It is currently anticipated that the testing provisions incorporated in the generic Westinghouse design will be incorporated in the Indian Point Unit No. 2 plant specific design, to be submitted to NRC for pre-implementation review as noted in 4.3 above. Included with that package will be a description of the design testing provisions.

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

Response to Item 2

The Indian Point Unit No. 2 reactor trip system is designed to permit periodic on-line testing. Accordingly, this item requires no further action with respect to Indian Point Unit No. 2.

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

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

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

Response to Item 3

Most recently, in light of the Salem event, Con Edison has reviewed the existing surveillance test program, its frequencies, and the fact that there have been no recorded failures of the breakers to open on demand. Based on this review Con Edison continues to consider the program acceptable.

The Westinghouse Owners Group has contracted with Westinghouse for the development of a methodology for evaluating surveillance frequencies and out of service times for the reactor protection instrumentation system. On the basis of the results of this Westinghouse study and the NRC Safety

Evaluation Report, Technical Specification changes to the Reactor Trip System equipment surveillance test frequencies may be proposed, if warranted.

TABLE 1.2-1
SEQUENCE OF EVENTS - DIGITAL INPUTS

INPUT EVENTS

RC Loop A Low Flow - causing Reactor Trip
RC Loop B Low Flow - causing Reactor Trip
RC Loop C Low Flow - causing Reactor Trip
RC Loop D Low Flow - causing Reactor Trip
Steam Line Hi Flow, SI, - causing Reactor Trip
SG - A Lo-Lo Level, - causing Reactor Trip
SG - B Lo-Lo Level, - causing Reactor Trip
SG - C Lo-Lo Level, - causing Reactor Trip
SG - D Lo-Lo Level, - causing Reactor Trip
Pressurizer Hi-Level, - causing Reactor Trip
PWR Range Chan, Hi-Flux, - causing Reactor Trip
PWR Range Chan, Lo-Flux, - causing Reactor Trip
Intermed. Rnge 1, Hi-Flux, - causing Reactor Trip
Intermed. Rnge 2, Hi-Flux, - causing Reactor Trip
Source Rnge 1, Hi-Flux, - causing Reactor Trip
Source Rnge 2, Hi-Flux, - causing Reactor Trip
Turbine Hyd. Oil, Lo-Press., - causing Reactor Trip
Stm, Line A, Hi. Diff. Press., SI, - causing Reactor Trip
Stm, Line B, Hi. Diff. Press., SI, - causing Reactor Trip
Stm, Line C, Hi. Diff. Press., SI, - causing Reactor Trip
Stm, Line D, Hi. Diff. Press., SI, - causing Reactor Trip
Pressurizer, Hi-Press., - causing Reactor Trip
Pressurizer, Lo-Press., - causing Reactor Trip
Containment, Hi-Press, - causing Reactor Trip
RC Overtemp. delta - T, - causing Reactor Trip
RC Over Power delta - T, - causing Reactor Trip
RC Pump Bus Under Voltage, - causing Reactor Trip
Reactor Manual Trip 1, - causing Reactor Trip
Reactor Manual Trip 2, - causing Reactor Trip
Reactor Main Trip Breaker A, - causing Reactor Trip
Reactor Main Trip Breaker B, - causing Reactor Trip
Reactor Aux Trip Breaker A, - causing Reactor Trip
Reactor Aux Trip Breaker B, - causing Reactor Trip
Unit On-Line Tie Breaker, - causing Reactor Trip
Reactor Coolant Pump A Breaker Open'g, - causing Reactor Trip
Reactor Coolant Pump B Breaker Open'g, - causing Reactor Trip
SG-A, Lo-Level and Lo-FW Flow, - causing Reactor Trip SG-B, Lo-Level and Lo-FW Flow, - causing Reactor Trip
SG-C, Lo-Level and Lo-FW Flow, - causing Reactor Trip
SG-D, Lo-Level and Lo-FW Flow, - causing Reactor Trip
Reactor Coolant Pump C Breaker Op'g - causing Reactor Trip

Table 1.2-1 (con't)
SEQUENCE OF EVENTS
DIGITAL INPUTS

Reactor Coolant Pump D Breaker Op'g - causing Reactor Trip
Pressurizer, Lo-Press, SI, - causing Reactor Trip
Safety Inj Manual Set 1, - causing Reactor Trip
Safety Inj Manual Set 2, - causing Reactor Trip
RC Overpress. Prot. Sys Train A or B Operated
Low Turb. Br. Oil Press, - causing Turb Trip
Low Condenser vacuum, - causing Turbine Trip
High Vibration, - causing Turbine Trip
Gen. 86BU Lockout Relay, - causing Turbine Trip
Gen. 86 P Lockout Relay, - causing Turbine Trip
Mech. Overspeed, - causing Turbine Trip
Thrust Bearing Wear, - causing Turbine Trip
CCR Manual Trip - causing Turbine Trip
Redundant Overspeed A, - causing Turbine Trip
Redundant Overspeed B, - causing Turbine Trip

Table 1.2-2

SAMPLE - SEQUENCE OF EVENTS REPORT: INDIAN POINT 2

10:14:08:4 SEQUENCE OF EVENTS TRIP TIME - 10:13:07:6 STATUS Elapsed MSEC
INDIAN POINT UNIT 2 - 06 AUG 83

SEQ. OF EVTS.	PD1003	CONTAINM HI P SI CAUS RE*	TR	0.
SEQ. OF EVTS.	YD0007	REAC MAIN TR BKR B	TR	163.
SEQ. OF EVTS.	YD9039	GEN 86BU LOCKOUT RELAY CAUS TU**	TR	165.
SEQ. OF EVTS.	YD9040	GEN 86P LOCKOUT RELAY	TRIP	165.
SEQ. OF EVTS.	YD0006	REAC MAIN TR BKR A	TR	169
SEQ OF EVTS.	PD0399	TB HYD OIL LO P CAUS RE	TR	309.
SEQ. OF EVTS.	YD9044	REDUNDANT OVERSPEED B CAUS TUR	TRIP	873.
SEQ. OF EVTS.	YD9045	REDUNDANT OVERSPEED A CAUS TUR	TRIP	886.
SEQ. OF EVTS.	TD0499	RCL OVERPWR DT CAUS RE	TR	1074.
SEQ. OF EVTS.	TD0498	RCL OVERTEMP DT CAUS RE	TR	1075.
SEQ. OF EVTS.	ND0020	INTERM RNG 1 HI Q CAUS RE	NOT TR	1084.
SEQ. OF EVTS.	ND0021	INTERM RNG 2 HI Q CAUS RE	NOT TR	1094.
SEQ. OF EVTS.	ND0010	PWR RNG CHAN LO Q CAUS RE	NOT TR	1208.
SEQ. OF EVTS.	TD0490	RCL OVERTEMP DT CAUS RE	NOT TR	1323.
SEQ. OF EVTS.	TD0499	RCL OVERPWR DT CAUS RE	NOT TR	1730.
SEQ. OF EVTS.	YD0421	STM GEN B LO L & FW F CAUS RE	TR	2855.
SEQ. OF EVTS.	YD0401	STM GEN A LO L & FW F CAUS RE	TR	2928.
SEQ. OF EVTS.	YD0441	STM GEN C LO L & FW F CAUS RE	TR	3670.
SEQ. OF EVTS.	YD0461	STM GEN D LO L & FW F CAUS RE	TR	3977.
SEQ. OF EVTS.	LD0406	STM GEN A LO LO L CAUS RE	TR	5887.
SEQ. OF EVTS.	LD0426	STM GEN C LO LO L CAUS RE	TR	5922.
SEQ. OF EVTS.	LD0446	STM GEN C LO LO L CAUS RE	TR	6169.

END OF SEQUENCE OF EVENTS PRINTOUT

Footnotes: * - "RE" - Reactor Trip
** - "TU" - Turbine Trip

TABLE 1.2-3
TRIP LOGS ANALOG INPUTS

Logs 1 & 5 Sampling Rate of 5 Sec

Power Range Channel & Neutron Detectors
Source Range Detectors
Intermediate Range Detectors
Pressurizer Level
Steam Generator A Narrow Range Level
Steam Generator B Narrow Range Level
Steam Generator C Narrow Range Level
Steam Generator D Narrow Range Level
Steam Generator A Wide Range Level
Steam Generator B Wide Range Level
Steam Generator C Wide Range Level
Steam Generator D Wide Range Level
RCS Loop A Flow
RCS Loop B Flow
RCS Loop C Flow
RCS Loop D Flow

Logs 2 & 6 Sampling Rate of 5 Sec

Steam Generator A Pressure
Steam Generator B Pressure
Steam Generator C Pressure
Steam Generator D Pressure
Steam Generator A Feedwater In
Steam Generator B Feedwater In
Steam Generator C Feedwater In
Steam Generator D Feedwater In
Steam Generator A Feedwater Out
Steam Generator B Feedwater Out
Steam Generator C Feedwater Out
Steam Generator D Feedwater Out
Core Delta Flux
Steam Generators Corrected Average Feedwater flow
Steam Generators Uncorrected Average Steam flow
Steam Generators Steam Pressure

Logs 3 & 7 Sampling Rate of 10 Sec

Reactor Coolant Loop T - average (each loop)
Reactor Coolant Loop Delta T
Reactor Coolant Loop T - Reference
Reactor Coolant Loop Average Delta T
Reactor Coolant Loop Average T Average
Pressurizer Water Temperature
Pressurizer Steam Temperature
Presssurizer Surge Line Temperature
Pressurizer Spray Water Inlet Temperature

Pressurizer Relief Tank Temperature
Reactor Coolant Loop Cold Leg Temperature (each loop)
Reactor Coolant Loop Overpower delta T setpoint (each loop)
Reactor Coolant Loop Over temperature delta T setpoint (each loop)
Reactor Coolant Pump Motor A bearing temp
Reactor Coolant Pump Motor B bearing temp
Reactor Coolant Pump Motor C bearing temp
Reactor Coolant Pump Motor D bearing temp

Logs 4 & 8 Sampling Rate of 10 Sec

Reactor Coolant Loop System Pressure
Pressurizer Pressure
Pressurizer Relief Tank Pressure
Containment Pressure

Note: Sampling Period (Duration of Time History for Trip)=Sampling Rate x 40

INDIAN POINT 2 LOG 3 PRETRP 3 PRIMARY SYSTEM TEMPERATURES 10:08:27 FREQ 10 SECS

DEMAND TRIP LOG AT 10:03:16 ON 04 NOV 83

GROUP NO 8 TAVG AND DELTA T 04 NOV 83 PAGE 1

TIME
1 RCLA 1 TAVG (NR)
2 RCLB 1 TAVG (NR)
3 RCLC 1 TAVG (NR)
4 RCLD 1 TAVG (NR)
5 RCLA 1 DT
6 RCLB 1 DT
7 RCLC 1 DT
8 RCLD 1 DT
9 RC TREF (NR)
10 RCL AVG DT
11 RCL AVG TAVG (NR)
12
13

TIME	1 T0400 DEGF	2 T0420 DEGF	3 T0440 DEGF	4 T0460 DEGF	5 T0403 DEGF	6 T0423 DEGF	7 T0443 DEGF	8 T0463 DEGF	9 T0496 DEGF	10 T0497 DEGF	11 T0499 DEGF	12	13
09:51:16	XXXXXXB	549.1	549.0	546.1	-14.3	55.5	52.7	49.4	549.3	52.9	550.2		
09:51:26	XXXXXXB	549.0	548.8	546.4	-14.3	55.3	52.6	49.9	549.2	53.1	551.0		
09:51:36	XXXXXXB	549.1	549.3	546.6	-14.3	55.6	52.9	50.3	549.3	53.3	550.6		
09:51:46	XXXXXXB	549.1	549.0	546.1	-14.3	55.3	52.4	48.9	549.3	52.7	549.6		
09:51:56	XXXXXXB	548.8	549.2	546.0	-14.3	55.1	52.6	49.3	549.2	52.8	550.4		
09:52:06	XXXXXXB	548.8	549.3	546.4	-14.3	55.3	53.1	50.4	549.2	53.1	551.2		
09:52:16	XXXXXXB	549.0	549.0	546.6	-14.3	55.4	52.5	49.8	549.3	52.9	550.0		
09:57:13	XXXXXXB	548.9	548.7	546.2	-14.3	55.2	52.5	49.2	549.3	52.7	550.1		
09:57:23	XXXXXXB	548.9	548.7	546.2	-14.3	55.2	52.5	49.2	549.3	52.7	550.1		
09:57:33	XXXXXXB	548.9	548.7	546.2	-14.3	55.2	52.5	49.2	549.3	52.7	550.1		
09:57:43	XXXXXXB	548.9	548.7	546.2	-14.3	55.2	52.5	49.2	549.3	52.7	550.1		
09:57:54	XXXXXXB	548.8	548.8	546.1	-14.3	55.2	52.8	49.7	549.2	52.8	549.9		
09:58:12	XXXXXXB	549.0	548.6	545.9	-14.3	55.4	52.5	48.9	549.3	52.6	549.5		
09:58:26	XXXXXXB	548.8	548.9	546.3	-14.3	55.2	52.9	50.1	549.3	53.0	550.2		
09:58:40	XXXXXXB	548.9	548.7	546.1	-14.3	55.5	52.3	49.3	549.3	52.9	549.1		
09:58:55	XXXXXXB	548.9	548.9	545.9	-14.3	55.3	52.7	49.3	549.3	52.9	550.7		
09:59:08	XXXXXXB	549.0	548.9	546.4	-14.3	55.6	52.5	49.7	549.3	53.0	550.1		
09:59:21	XXXXXXB	548.9	548.9	546.4	-14.3	55.3	52.4	50.3	549.3	53.0	549.9		
09:59:33	XXXXXXB	548.9	548.9	546.4	-14.3	55.2	52.9	50.1	549.3	53.1	550.4		
09:59:42	XXXXXXB	548.8	548.9	546.2	-14.2	55.4	53.1	49.9	549.3	53.1	549.8		
09:59:53	XXXXXXB	548.8	548.8	546.3	-14.3	55.4	53.0	50.1	549.3	53.1	549.8		
10:00:03	XXXXXXB	548.7	549.2	545.9	-14.3	55.4	53.5	49.8	549.3	53.1	550.2		
10:00:18	XXXXXXB	548.7	548.9	546.2	-14.3	55.4	53.2	50.2	549.3	53.1	550.7		
10:00:27	XXXXXXB	548.8	548.4	546.2	-14.2	55.5	52.6	49.9	549.3	53.2	550.2		
10:00:39	550.7	548.8	548.2	546.2	XXXXXXB	55.3	52.0	49.7	549.3	52.7	549.4		
10:00:46	558.2	548.6	546.9	545.8	XXXXXXB	55.3	53.0	49.1	549.3	52.9	550.0		
10:00:57	XXXXXXB	548.9	548.9	546.0	XXXXXXB	55.5	53.2	49.4	549.2	53.0	549.8		
10:01:09	XXXXXXB	549.0	548.8	546.0	XXXXXXB	55.6	52.8	49.3	549.3	52.8	549.6		
10:01:22	XXXXXXB	548.9	548.9	545.8	XXXXXXB	55.8	52.7	49.1	549.3	52.9	549.7		
10:01:32	XXXXXXB	549.1	549.0	545.8	XXXXXXB	55.9	52.8	49.0	549.3	53.1	550.4		
10:01:43	595.1	548.9	548.6	546.2	-13.3	55.5	52.0	50.1	549.3	52.9	550.3		
10:01:57	579.2	549.0	548.8	546.5	-2.1	55.6	52.8	50.1	549.3	53.1	550.7		
10:02:07	560.0	549.1	548.8	545.9	39.4	55.8	52.8	49.6	549.2	53.2	550.4		
10:02:17	560.0	549.0	548.7	546.0	39.0	55.7	52.6	49.0	549.3	52.9	549.5		
10:02:28	559.3	549.0	548.9	545.9	40.5	55.5	52.7	49.1	549.2	52.9	550.3		
10:02:38	559.6	548.9	549.5	545.9	40.0	55.7	53.3	49.2	549.3	53.0	549.2		
10:02:47	559.6	548.9	546.9	546.3	39.9	55.6	52.5	50.1	549.3	53.1	550.3		
10:02:57	559.6	549.0	548.9	546.0	39.8	55.6	52.7	49.7	549.2	52.9	549.7		
10:03:09	559.6	548.9	548.8	545.8	39.9	55.3	52.8	49.3	549.3	52.8	550.0		
10:03:16	559.6	548.9	548.8	546.2	39.9	55.5	52.5	50.1	549.3	53.0	550.0		

INDIAN POIN[illegible]T 2 LOG 7 POSTRP 3 PRIMARY SYSTEM TE[illegible]TURES 10:13:26 FREQ 10 SECS

DEMAND TRIP LOG AT 10:03:16 ON 04 NOV 83

GROUP NO 8 TAVG AND DELTA T 04 NOV 83 PAGE 1

TIME
1 RCLA 1 TAVG (NR)
2 RCLB 1 TAVG (NR)
3 RCLC 1 TAVG (NR)
4 RCLD 1 TAVG (NR)
5 RCLA 1 DT
6 RCLB 1 DT
7 RCLC 1 DT
8 RCLD 1 DT
9 RC TREF (NR)
10 RCL AVG DT
11 RCL AVG TAVG (NR)
12
13

TIME	1 T0400 DEGF	2 T0420 DEGF	3 T0440 DEGF	4 T0460 DEGF	5 T0403 DEGF	6 T0423 DEGF	7 T0443 DEGF	8 T0463 DEGF	9 T0496 DEGF	10 T0497 DEGF	11 T0499 DEGF	12	13
10:03:20	559.7	548.9	549.1	546.4	39.9	55.4	53.1	50.3	549.3	53.2	550.5		
10:03:34	559.6	548.8	548.6	546.4	39.9	55.2	52.7	50.1	549.3	53.2	550.1		
10:03:47	559.6	549.0	548.9	546.2	39.9	55.4	52.8	49.4	549.3	53.0	550.4		
10:03:57	559.6	549.0	549.0	546.1	39.9	55.6	52.5	49.2	549.2	53.0	550.0		
10:04:08	559.6	548.9	548.9	546.0	39.9	55.6	52.5	49.4	549.3	52.9	550.2		
10:04:17	559.6	548.8	548.9	546.4	39.9	55.3	52.8	49.8	549.3	52.9	551.1		
10:04:28	559.7	548.7	549.1	545.9	39.9	55.0	53.2	49.2	549.3	52.7	549.5		
10:04:37	559.6	549.1	548.7	546.4	39.9	55.6	52.7	50.1	549.2	53.2	550.8		
10:04:47	559.6	548.8	548.8	546.3	39.9	55.2	52.5	50.2	549.3	52.8	551.0		
10:04:57	559.6	548.9	549.1	546.6	39.9	55.5	53.1	50.2	549.3	53.3	551.5		
10:05:08	559.6	548.9	549.0	546.5	39.9	55.3	52.8	50.2	549.2	53.1	550.0		
10:05:18	559.6	548.8	548.9	546.2	39.9	55.3	52.8	49.7	549.2	53.1	550.0		
10:05:28	559.6	548.9	549.1	546.0	39.9	55.5	52.9	49.2	549.3	52.9	549.4		
10:05:38	559.6	548.9	549.0	546.4	39.9	55.3	52.7	50.1	549.2	53.1	550.4		
10:05:46	559.6	548.9	548.9	546.5	39.9	55.5	52.4	50.3	549.2	53.1	550.4		
10:05:58	559.6	548.9	548.6	546.0	39.9	55.4	52.4	49.7	549.3	52.8	550.1		
10:06:07	559.6	549.1	549.2	546.3	39.9	55.8	52.8	49.9	549.3	53.3	550.8		
10:06:17	559.6	548.8	548.8	546.2	39.9	55.2	52.2	49.8	549.3	52.8	550.5		
10:06:27	559.7	549.1	548.8	546.4	39.9	55.5	52.5	50.4	549.2	53.1	550.1		
10:06:38	559.6	549.1	549.0	546.0	39.9	55.8	52.6	50.1	549.2	53.1	549.9		
10:06:46	559.7	548.7	549.1	546.0	39.9	55.2	52.5	49.0	549.3	53.0	551.1		
10:06:56	559.7	548.9	549.0	546.1	39.9	55.3	52.3	49.0	549.3	52.8	549.7		
10:07:07	559.6	549.0	548.9	546.1	39.9	55.4	52.7	49.2	549.3	52.8	550.7		
10:07:16	559.6	549.0	548.8	546.4	39.9	55.3	52.8	49.9	549.2	53.1	550.6		
10:07:26	559.6	549.1	548.9	545.9	39.9	55.6	52.8	49.1	549.2	53.0	550.3		
10:07:36	559.6	548.9	548.7	546.3	39.9	55.5	52.5	49.6	549.3	52.9	549.9		
10:07:46	559.6	549.1	548.7	546.5	39.9	56.0	52.9	50.2	549.3	53.3	550.1		
10:07:56	559.6	548.9	548.8	545.9	39.9	55.5	52.6	49.2	549.3	52.9	549.9		
10:08:06	559.6	548.9	548.8	545.9	39.9	55.4	52.1	49.4	549.3	52.9	550.3		
10:08:16	559.7	549.0	548.8	546.5	39.9	55.6	52.6	50.1	549.2	53.2	550.7		
10:08:26	559.6	549.1	548.6	546.2	39.9	55.8	51.4	49.3	549.3	52.8	549.4		
10:08:36	559.6	549.0	548.7	545.9	39.9	55.5	51.9	49.0	549.3	52.8	550.7		
10:08:46	559.6	549.1	548.5	545.9	39.9	55.6	51.8	49.1	549.3	52.7	549.7		
10:08:56	559.6	548.9	548.5	546.3	39.9	55.4	51.7	50.3	549.2	52.8	550.3		
10:09:06	559.7	549.0	548.4	546.5	39.9	55.7	51.5	50.1	549.3	52.9	550.0		
10:09:16	559.6	548.9	548.6	546.5	39.9	55.4	52.1	50.0	549.3	52.9	550.1		
10:09:26	559.6	548.8	549.1	546.2	39.9	55.0	52.7	50.0	549.3	52.8	550.4		
10:09:36	559.6	549.1	549.2	546.0	39.9	55.6	53.1	49.1	549.3	52.9	549.3		
10:09:46	559.7	548.9	549.3	546.1	39.9	55.4	53.4	49.0	549.2	53.0	549.8		
10:09:56	559.6	548.8	549.1	545.9	39.9	55.2	52.6	48.9	549.3	52.7	549.6		

Table 1.2-5 (Page 1 of 4)
CONTROL ROOM RECORDERS

DESCRIPTION	CHART SPEED	RANGE OF INST.	LOCATION
Fan Cooler Weir Lv.	2 Cm/hr	1-5v=0-45" 3 Pens	SO PNL
Fan Cooler Weir Lv.	2 Cm/hr	1-5v=0-45" 3 Pens	SO PNL
Dew Pt. F.C.U.	1"/hr	30-120°F 5 Points	SO PNL
Containment Press Air Flow	1"/hr	0-15 CFM	SL PNL
Weld Ch. Press.	1"/hr 3"/hr	0-60# 0-120# 12 Points	SL PNL
C.P.D. Sodium Flame Photometers	2 Cm/hr	0-20ppB Red 4-20 ma Blue 0-100MV	SK PNL
CL. P.P.B. 21-22 Circulator	2 Cm/hr	4-20 ma=0-50 PPB	SK PNL
CL. P.P.B. 23-24 Circulator	2/Cm hr	4-20 ma 0-50 PPB	SK PNL
C.L P.P.B. 25-26 Circulator	2/Cm hr	4-20 ma 0-50 PPB	SK PNL
Residual HXIN HXOUT	1"/hr	75-400°F	SG PNL
R.C.S. Press Recorder	1"/hr 15"/hr	0-750# 10-20 ma	SF PNL
Turbine Startup Temp	1"/hr 3"/hr	0-600°F 10-Points	SE PNL
Turbine Supervisory	1"/hr 3"/hr	24 Points 0-15 Mills 0-2 Mills 0-1 Mills	SE PNL
Turbine Rotor Position	4"/hr	0-120 Mills	SE PNL
Vlv % Speed R.P.M.	4"/hr	0-100% 0-2500 RPM	SE PNL
Secondary Press	1"/hr 3"/hr	0-300# 0-1200# 12 Points	SD PNL
Secondary Plnt Temp	1"/hr 3"/hr	0-600°F SD 17 Points	SD PNL
Condenser Press	24 Hrs	0-30" HG	SD PNL
#26 Heater Inlet/Outlet	1"/hr	0-500°F	SC
21/22 BFP Reheat S.F.	1"/hr	0-120#/hr	SC
2A/2B Reheat Drn Tnk F	1"/hr	0-90gpm	SC
Avg Temp 21, 22, 23 A&B	1"/hr	0-600°F (2 Pens)	SC
21/22 BFP Suction Flow	1"/hr	0-180x100 GPM	SC
21/22 Stm. Gen. Level	1"/hr	0-100%	SC
23/24 Stm. Gen. Level	1"/hr	0-100%	SC

Table 1.2-5 (Page 2 of 4)

Description	Chart Speed	Range of Inst	Location	
R13 Plant Vent Part	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R14 Plant Vent Gas	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R16 F.C.U. Liquid	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R11 V.C. Particulate	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R12 V.C. Radiogas	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R23 FCU Liquid	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R19 Stm. Gen. Blow DN	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R17 CCW Liquid	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R15 Air Ejector Radio Gas	3"/hr	10^{-1}-10^{-6} CPM	SA1	
R18 Liquid Waste	3"/hr	10^{-2}-10^{-6} CPM	SA1	
R20 Waste Gas	3"/hr	10-10^{-4} CPM	SA1	
Boric Acid Flow	1"/hr	0-10gpm	FD	
Pri Water Flow		0-10^{12}gpm	FD	
Press Level S.P.	1"/hr	0-100%	FD	
Press Level		0-100%	FD	
Press Press	1"/hr	1700-2500#	FD	
OT delta t SP	1"/hr	0-75°F	FD	
Loop delta t		0-75°F	FD	
OP delta t SP		0-75°F	FD	
T-Ref	1"/hr	540-615°F	FD	
Avg T-Avg			FD	
T Avg	1"/hr	544-552°F		
delta t	1"/hr	-3.75-0+3.75°F	FD	
#1 Seal Return Flow	1"/hr	0-1 gpm	FD	
#23-#24 RCP			FD	
#1 Seal Return Flow	1"/hr	0-1 gpm	FD	
#22-#21 RCP				
Loop 24	1"/hr	0-3000#	FD	
Hot Leg Press				
#21-22 RC Loop Temp °F	1"/hr	0-600°F	FD	
#23-24 RC Loop Temp °F	1"/hr	0-600°F	FD	
Seal Return Hi Range	1"/hr	0-6 gpm	FD	#23#24
Seal Return Hi Range	1"/hr	0-6 gpm	FD	#22#21
NIS Overpower CH#1&3	1"/hr	0-200%	FDF	
NIS OverPower Ch#2&4	1"/hr	0-200%	FDF	
NIS Recorder		0-120%	FCF	
		SR 10^{-11} to 10^{-4}amos		
		INT RG 10^{0} to 10^{6}cpm		
21-Feed Flow	1"/hr	0-4x10^{6} gpm #/hr	FBF	
Steam Flow	1"/hr	0-4x10^{6} gpm #/hr		
St. Gen Level	1"/hr	0-100%		
22-Feed Flow	1"/hr	0-4x10^{6} gpm #/hr	FBF	
Steam Flow	1"/hr	0-4x10^{6} gpm #/hr		
St. Gen. Level	1"/hr	0-100%		
23-Feed Flow	1"/hr	0-4x10^{6} gpm #/hr	FBF	
Steam Flow	1"/hr	0-4x10^{6} gpm #/hr		
St. Gen. Level	1"/hr	0-100%		
24-Feed Flow	1"/hr	0-4x10^{6} gpm #/hr	FBF	
Steam Flow	1"/hr	0-4x10^{6} gpm #/hr		
St. Gen. Level	1"/hr	0-100%		

Table 1.2-5 (Page 3 of 4)

Description	Chart Speed	Range	Location
Control Rod Bank Insertion Limits	1"/hr	0-240 Steps 8 Points	FDR
RCP Winding Temp	1"/hr	0-300°F 4 Points	FDR
RAD Iodine R1-1	1"/hr	$10-10^6$ cpm	D2
RAD Iodine R1-2, R1-3, R1-4	1"/hr	$10-10^6$ cpm	D2
Flux Map A-B	7.2"/Min	0-10 MV	D7
Flux Map C-D	7.2"/Min	0-10 MV	D5
Flux Map E-F	7.2"/Min	0-10 MV	D5
RCP Vibration	1"/hr	0-20 Mills Vib 8 Points	C9
Gross Failed Fuel	1" or 6"/hr	$10-10^6$ cpm	C9
Cond. River H_2O Temp	2 CM/hr	90 to 100°F, -10° to +15°F	Assmnt. PNL (Rear Unit 1)
	2 CM/hr	50-100°,0-40°F, 2-12°F	
	1"/hr	25°-85°F	Assmnt. PNL(rear)
Cond. St Tank Levels	24 hrs	0-42%	Assmnt. PNL(rear)
Fresh H_2O Cooling Temp	24 hrs	0-400°F	Assmnt. PNL(rear)
Primary Coolant Temp	24 hrs	0-700°F	Assmnt. PNL(rear)
Primary Coolant Inlet Temp 11&12	24 hrs	0-700°F	Assmnt. PNL(rear)
Primary Coolant Inlet Temp 13&14	24 hrs	0-700°F	Assmnt. PNL(rear)
Primary Coolant Flow delta P	24 hours	0-750 #	Assmnt. PNL(rear)
Clean H_2O Storage Tk	24 hrs	0-300"	Assmnt. PNL(rear)
Radiogas - Annulus Air Particulate	3"/hr	0-150 CPS	Assmnt. PNL(rear)
Sphere Foundation Drain Sump Radiation	3"/hr	0-1000 CPS	Assmnt. PNL(rear)
Sewage Holdup Tank Radiation	3"/hr	0-1000 CPS	Assmnt. PNL(rear)
Air Particulate	3"/hr	0-1000 CPS	Assmnt. PNL(rear)
Stack Particulate			Assmnt. PNL(rear)
Containment Cooling	3"/hr	0-1000 CPS	Assmnt. PNL(rear)
Blowdown Effluent	3"/hr	0-1000 CPS	Assmnt. PNL(rear)
Gen. Stator Temp	3"/hr	0-150°C 29 Points	Assmnt. PNL(rear)
Wind Recorder	3"/hr	0-180° 0-44.8 m/s	Assmnt. PNL(side)
Salinity	3"/hr	0-20 PPB Nacl 0-200 PPB Nacl	Assmnt. PNL
Mainsteam Line 21, 22, 23, 24 Hi RAD.			Assmnt. PNL #1
RCS Hot Leg St. Gen 22&24	1"/hr	0-700°F	Assmnt. PNL #2
VC Sump	1"/hr		
Hi Range Press		0-150 PSI	A.P. #2

Table 1.2-5 (Page 4 of 4)

Description	Chart Speed	Range	Location
Cont. Sump		38'8.5"-56'8.5"	
RE-26	3"/hr	$10^{0}-10^{7}$ RADS/hr	Assmnt PNL #2
Toxic Gas Monitor	3"/hr	0-50 PPM	Assmnt PNL #1
Toxic Gas Monitor	3"/hr	0-50 PPM	Assmnt PNL #2
H2/O2 Analyzer	3"/hr	0-25% O_2 & 0-10%H_2	Assmnt PNL #2
RCS Hot Leg Temp S. G. 21&23	1"/hr	0-700°F	Assmnt PNL #3
R27 Vent	2cm/hr	$10^{+5}-10^{-7}$ μCi/cc	Assmnt PNL #3
R27 Stack	2cm/hr	$10-10^{13}$ μCi/sec	Assmnt PNL #3
Recirc & Press Sump Level & Press	1"/hr	34'10-7/8"-52'10-7/8"	Assmnt PNL #3
		0-150 psi	Assmnt PNL #3
RE25	3" hr	$10^{0}-10^{7}$ Rad/hr	Assmnt PNL #3
RCS Level Narrow	1" hr	0-110%	Assmnt PNL #3
RCS Level Wide			Assmnt PNL #3
H_2&O_2 Analyzer	3"/hr	0-25% O_2	Assmnt PNL #3
		0-10%H_2	Assmnt PNL #3

COL-1A

Electrical Relay Positions Following a Trip and Prior to Startup

Check Off List

COL Completed	Trip	Startup
Date		
Time		

Comments: _____

SWS _____

Jack C. Rodolf
Reviewed

M. J. Finck #566 5/25/82
SNSC Review / Date

J. Cooney 6/5/82
Approved / Date

Reference Drawings: None

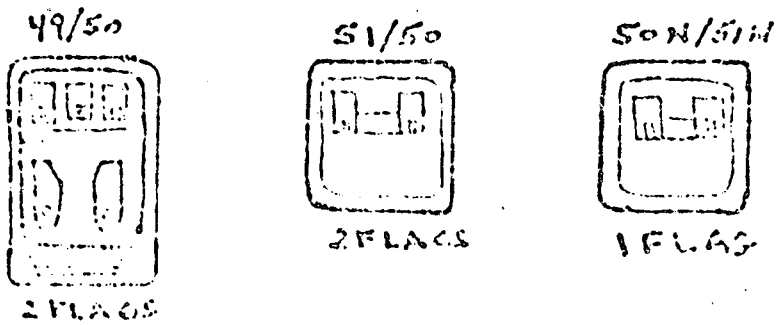
NOTE: 1. This COL should be done immediately following a unit trip and completed prior to the reactor going critical upon recovering from the trip.

2. Use a check to indicate those relay flags which are up. For relays with more than one flag additional lines are provided in the same pattern as flags exist in relay, looking left to right. An example is given below:

3. Complete the relay operations report.

Example: Section 3.6 23 Circulator		Trip		Inits.	Startup	Inits.
49/50	✓	✓		JCR		
51/50 Overcurrent Phase/Phase		✓	✓	JCR		
51N/50N Overcurrent Ground			✓	JCR		

Looking at the relays you would see:



Electrical Relay Positions

	Trip	Initials		Startup	Initi.	
1.0 Central Control Room						
1.1 Back of Unit II Flight Panel FAF						
TR 1 Target B.U. Direct Trip from Buchanan						
TR 1 Target Primary Direct Trip from Buchanan						
87/T21 A Ø Main Trans. Diff.						
87/T21 B Ø Main Trans. Diff.						
87/T21 C Ø Main Trans. Diff.						
87/T22 A Ø Main Trans. Diff.						
87/T22 B Ø Main Trans. Diff.						
87/T22 C Ø Main Trans. Diff.						
87/GT A Ø Overall Diff.						
87/GT B Ø Overall Diff.						
87/GT C Ø Overall Diff.						
87/UT A Ø UAT Diff.						
87/UT B Ø UAT Diff.						
87/UT C Ø UAT Diff.						
87/G Gen. Diff.						
40 Loss of Field						
60A Voltage Balance						
51/NT Main Trans. Neut. O.C.						
51N/UT Neut. O.C.						
59N Gen. Neut. O.C.						
46 Neg. Sequence						
62/UT Unit Aux. B.U. Timer						
51/UT A Ø Overcurrent						
51/UT B Ø Overcurrent						
51/UT C Ø Overcurrent						
Thrust Bearing Drum Level						
1.2 Back of Unit II Flight Panel FCF						
87 ST A Ø SAT. Diff.						
87 ST B Ø SAT. Diff.						
87 ST C Ø SAT. Diff.						
87 L2/345 P.W. Diff.						
32 NBU/345 B.U. Diff. Grd.						
87 L1/345 P.W. Diff.						
51 ST A Ø SAT. O.C.						
51 ST B Ø SAT. O.C.						
51 ST C Ø SAT. O.C.						
85L2/345 PWM & TT						
85L1/345 PWM & TT						
Station Aux. Trans. Neut. O.C.						
32 NP Pri. Dir. Grd.						
50 BU/345 BU & Fault						
50NP/345 Pri. Ø Fault						
50NP/345 Pri. Grd. Fault						

		Trip	Initials	Startup	Initi
62 STP	Sta. Trans. B.U. Timer				
62 T1-BT4-5	Bus 5 B.U. Timer				

1.3 Rear of Flight Panel FDF

87 L2/138	P.W. Diff.				
87 L1/138	P.W. Diff.				
85 L2/138	PWM & T.T.				
85 L1/138	PWM & T.T				
138KV Line	B.U. P.H. Fault Det.				
138KV Line	PRIM P.H. Fault Det.				
138KV Line	B.U. Ground Fault Det.				
138KV Line	Prim. Ground Fault Det.				
Unit Aux. Tap Changer Position		N		N	
Unit Aux. Tap Changer		OFF		OFF	
Sta. Aux. Tap Changer Position		N		N	
Sta. Aux. Tap Changer		OFF		OFF	

1.4 Power Failure System Sequence Cabinets

1.4.1 Cabinet #1

Stripping 27-1-5A				
Stripping 27-1-2A				
Stripping 27-2-6A				
Stripping 27-2-3A				
Black Out 27-51				
Black Out 27-52				
Black Out 27-53				

1.4.2 Cabinet #2

Stripping 27-2-5A				
Stripping 27-2-2A				
Stripping 27-1-6A				
Stripping 27-1-3A				
Black Out 27-61				
Black Out 27-62				
Black Out 27-63				

2.0 Exciter Breaker 33'Elev.

Dev. 76 T1X Trip, Dev. 76 T2X Trip
Dev. 27X Trip Damping Bias Trip

3.0 6.9 Switchgear Room 15'Elev.

3.1 SS5 - Station Service Trans.
51/50 Ø A Overcurrent
51/50 Ø B Overcurrent
51/50 Ø C Overcurrent

3.2 25 Circulator
49/50

51/50 Overcurrent Phase/Phase
51N/50N Overcurrent Ground

3.3 ST5 - Station Trans. Supply
62 ST5
51 ST5 Ø A
51 ST5 Ø B
51 ST5 Ø C
51N ST5

3.4 Aux. Unit 2
27-2A
27-2
81-2

3.5 SS2 - Station Service Trans.
51/50 Ø A Overcurrent
51/50 Ø B Overcurrent
51/50 Ø C Overcurrent

3.6 23 Circulator
49/50

51/50 Overcurrent Phase/Phase
51N/50N Overcurrent Ground

3.7 22 Condensate Pump
49/50

51/50 Overcurrent Phase/Phase
51N/50N Overcurrent Ground

3.8 24 RCP
49/50

51/50 Overcurrent Phase/Phase
51N/50N Overcurrent Ground

3.9 UT2 Unit Trans. Supply Bus 2
62 UT2
51 UT2 Ø A Overcurrent
51 UT2 Ø B Overcurrent
51 UT2 Ø C Overcurrent
51N UT2

		Trip	Initials		Startup	Initi.
3.10	UT1 Unit Trans Supply Bus 1					
	62 UT1					
	51 UT1 ϕ A Overcurrent					
	51 UT1 ϕ B Overcurrent					
	51 UT1 ϕ C Overcurrent					
	51 NUT1					
3.11	Aux. Unit 1					
	81-1					
	27-1					
	27-5					
	27-1A					
3.12	21 RCP					
	49/50					
	51/50 Overcurrent Phase/Phase					
	51N/50N Overcurrent Ground					
3.13	21 Condensate Pump					
	49/50					
	51/50 Overcurrent Phase/Phase					
	51N/50N Overcurrent Ground					
3.14	21 Circulator					
	49/50					
	51/50 Overcurrent Phase/Phase					
	51N/50N Overcurrent Ground					
3.15	21 Heater Drain Pump					
	49/50					
	51/50 Overcurrent Phase/Phase					
	51N/50N Overcurrent Ground					
3.16	26 Circulator					
	49/50					
	51/50 Overcurrent Phase/Phase					
	51N/50N Overcurrent Ground					
3.17	SS3 - Station Service Trans					
	51/50 ϕ A Overcurrent					
	51/50 ϕ B Overcurrent					
	51/50 ϕ C Overcurrent					

		Trip	Initials		Startup	Initials	
3.18	23 RCP						
	49/50	____	____	____	____	____	____
	51/50 Overcurrent Phase/Phase						____
	51N/50N Overcurrent Ground	____	____	____	____	____	____
		____	____	____	____	____	____
3.19	Aux. Unit 3						
	27-3A						
	27-3		____	____		____	____
	81-3		____	____		____	____
			____	____		____	____
3.20	UT-3 Unit Trans Supply Bus 3						
	62 UT3						
	51 UT3 Ø A Overcurrent		____	____		____	____
	51 UT3 Ø B Overcurrent		____	____		____	____
	51 UT3 Ø C Overcurrent		____	____		____	____
	51 NUT3		____	____		____	____
			____	____		____	____
3.21	UT-4 Unit Trans Supply Bus 4						
	62 UT4						
	51 UT4 Ø A Overcurrent		____	____		____	____
	51 UT4 Ø B Overcurrent		____	____		____	____
	51 UT4 Ø C Overcurrent		____	____		____	____
	51 NUT4		____	____		____	____
			____	____		____	____
3.22	22 RCP						
	49/50	____ ____	____	____	____	____	____
	51/50 Overcurrent Phase/Phase						____
	51N/50N Overcurrent Ground	____	____	____	____	____	____
		____	____	____	____	____	____
3.23	23 Condensate Pump						
	49/50	____ ____	____	____	____	____	____
	51/50 Overcurrent Phase/Phase						____
	51N/50N Overcurrent Ground	____	____	____	____	____	____
		____	____	____	____	____	____
3.24	24 Circulator						
	49/50	____ ____	____	____	____	____	____
	51/50 Overcurrent Phase/Phase						____
	51N/50N Overcurrent Ground	____	____	____	____	____	____
		____	____	____	____	____	____
3.25	22 Heater Drain Pump						
	49/50	____ ____	____	____	____	____	____
	51/50 Overcurrent Phase/Phase						____
	51N/50N Overcurrent Ground	____	____	____	____	____	____
		____	____	____	____	____	____

		Trip	Initials	Startup	Initials
3.26	Aux. Unit 4				
	81-4		____		____
	27-4		____		____
	27-6		____		____
	27-4A		____		____
3.27	ST6 Station Trans. Supply Bus 6				
	62 ST6		____		____
	51 ST6 ϕ A		____		____
	51 ST6 ϕ B		____		____
	51 ST6 ϕ C		____		____
	51 NST6		____		____
3.28	22 Circulator				
	49/50	____ ____	____	____	____
	51/50 Overcurrent Phase/Phase	____	____	____	____
	51N/50N Overcurrent Ground	____	____	____	____
3.29	SS6 Station Service Trans.				
	51/50 ϕ A Overcurrent	____	____	____	____
	51/50 ϕ B Overcurrent	____	____	____	____
	51/50 ϕ C Overcurrent	____	____	____	____
4.0	480 Volt Switchgear Room 15' Elev.				
4.1	EPG8				
	27S1/3A		____		____
	27S2/3A		____		____
4.2	EPG9				
	27S1/6A		____		____
	27S2/6A		____		____
4.3	EPG7				
	27S1/5A		____		____
	27S2/5A		____		____
4.4	EPG6				
	27S1/2A		____		____
	27S2/2A		____		____

RELAY OPERATIONS REPORT

[illegible] Power Generation
[illegible] No. 533.0

Station _____

Report No. _____

Date _____

Prepared by _____

REPORT ALL INFORMATION TO OPERATIONS CONTROL CENTER

Date of Trip-Out _____ Time _____ System at Fault _____ V/KV _____ Hertz

Fdr #, Gen., etc. _____ Bifurcated Fdr # _____

Fdr Grp/Assy _____ Bifurcated Fdr _____

Fdr Destination _____ Bifurcated Fdr _____

At Time of Trip: Gen: MW H.P. _____ L.P. _____ MVARS H.P. _____ Lead/Lag L.P. _____ Lead/Lag

Status of Generator Voltage Regulators: H.P. _____ On-Off L.P. _____ On-Off

Oscillograph Operation: Station _____ Time _____ Target _____

CIRCUIT OR APPARATUS INVOLVED	RELAY OPERATIONS				SKR OPER AUTO	REMARKS
	PRIMARY RELAY		AUXILIARY			
	TYPE	LIST TARGETS & PHASE OPERATED	TYPE	DESIGNATION		

CAUSE OF TROUBLE

LOCATION OF FAULT

Relay Operation Analyzed by Station I & C Supv.: Correct - Incorrect Name _____ Date _____

Reported to Central I & C: Date _____ Time _____

Restored to Service: Date _____ Time _____ Date (Bifurcated Fdr) _____ Time _____

GA-1206 7/30

Appendix A