Florida
Power
CORPORATION

July 31, 1984
3F0784-21

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Updated Response to Generic Letter 83-28

Dear Sir:

By letters dated July 8 and October 27, 1983, the NRC requested the status of Florida Power Corporation's (FPC's) current conformance with the positions contained in Generic Letter 83-28, and our plans and schedules for any needed improvements to conform with the positions. FPC evaluated our existing programs and procedures to establish our degree of conformance with the positions contained in the Generic Letter. FPC's Crystal River Unit 3 (CR-3) specific response to the Generic Letter was submitted on November 4, 1983. In addition to our internal efforts to address the Generic Letter, FPC participated in the B&W Owners Group's definition of generic efforts that could be undertaken on each of the Generic Letter's positions. The B&W Owners Group generic response was submitted on November 4, 1983. FPC also participated in the INPO sponsored Nuclear Utility Task Action Committee (NUTAC) formed to address Position 2.2.2. The NUTAC Final Report was issued on March 23, 1984.

Since the November 4, 1983, FPC response to the Generic Letter, significant information has become available and programs implemented which warrant an amendment to our original response. FPC hereby provides the enclosed updated CR-3 specific response to the entire Generic Letter. (NOTE: Revisions to the November 4, 1983, response are indicated by a vertical line in the right-hand margin. Attachments to the November 4, 1983, response have not been superseded unless otherwise noted and are not reproduced in this response.) The B&W Owners Group also amended its generic responses to Generic Letter Positions 4.1 and 4.2 on July 16, 1984.

A055 1/40 Site Aperture Card Dist
Extra Cy To: R Herman ORB#4

General Office 3201 Thirty-fourth Street South • P.O. Box 14042, St. Petersburg, Florida 33733 • 813—866-5151

Based on our evaluation of FPC's conformance with the positions contained in the Generic Letter, we have determined that existing FPC programs and procedures ensure a high degree of reactor trip system reliability.

If you have any questions, please contact this office.

Sincerely,

G. R. Westafer

G. R. Westafer
Manager, Nuclear Operations
Licensing and Fuel Management

DLT/feb

Enclosure

cc: Mr. James P. O'Reilly (Enclosure)
Regional Administrator, Region II
Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, GA 30323

STATE OF FLORIDA

COUNTY OF PINELLAS

G. R. Westafer states that he is the Manager, Nuclear Operations Licensing and Fuel Management, of Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

G R Westafer

G. R. Westafer

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 31st day of July, 1984.

Deborah Leonard

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: November 19, 1986

ENCLOSURE
UPDATED RESPONSE TO GENERIC LETTER 83-28
FOR CRYSTAL RIVER UNIT 3

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:

1. The criteria for determining the acceptabilty of restart.

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

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

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 Position 1.2).

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

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.

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

Response

Florida Power Corporation has in place and is maintaining a program to ensure that unplanned reactor shutdowns are analyzed and that a determination is made that the plant can be safely restarted. The following items address the programmatic elements.

1. The criteria for determining acceptability of restart is two-fold. Restart will be authorized <u>if</u> and only <u>if</u> safety systems respond as designed and the overall plant response was as designed. Additionally, restart will only be authorized upon satisfactory determination of the cause of the unplanned reactor shutdown.

2. The personnel performing restart analysis are the "person-on-call" and the Shift Operations Technical Advisor (STA).

   The "person-on-call" is a senior plant management individual holding a current Senior Reactor Operator License or equivalent and is the designee of the Plant Manager. The "person-on-call" is the evaluation team leader in assessing and justifying restart.

   The STA is an experienced plant response specialist. The STA gathers data from which the restart decision is de. The "person-on-call" and the STA jointly ensure that tne restart criteria are met.

3. The "person-on-call" must: 1) hold a current Senior Reactor Operator License or have acquired the experience and equivalent training normally required to be eligible for a Senior Reactor Operator's License whether or not the examination is taken, 2) have completed emergency coordinator training, and 3) be designated as the "person-on-call" by the Plant Manager.

   The STA must have completed Florida Power Corporation's STA Training Program and be designated as the STA by the Nuclear Safety Supervisor.

4. The STA and "person-on-call" use the following plant information sources in analyzing and reconstructing the event (refer to the response to Position 1.2 for details).

   Reactor Protection System Cabinets
   Annunciator Events Printouts
   Computer Alarm Printouts
   Post Trip Summary
   RECALL System

5. Event analysis methods used by Florida Power Corporation have been developed from the well-known Kepner-Tregoe (K-T) problems analysis approach. A form was developed and is in place which is used to document the event analysis.

   Additionally, Crystal River Unit 3 utilizes an event recall system. This system is used in comparing event information with expected plant behavior. For use with this system we have a library of expected plant transient behaviors.

6. Independence in evaluating the event is inherent in the evaluation team as discussed in items 2 and 3. Additionally, this team is the most competent group to perform the analysis.

   FPC routinely maintains records and logs to facilitate good operations and to enhance followup evaluations. See the response to Position 1.2 for data and information capability. The data and information used in the performance of an evaluation is attached to the report and kept on file.

7. The combination of the K-T method, the analysis documentation form, and the requirement to satisfactorily determine the cause of the event and take corrective action, as appropriate, comprise the systematic method for conducting restart evaluations. The process used to conduct restart evaluations is documented in the FPC Operations Section Implementation Manual, Section IV, Paragraph H. See Attachment 1 of our 11/4/83 response. R1

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 (Position 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

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

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

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

    2. Parameters monitored

    3. Time discrimination between events

    4. Format for displaying data and information

    5. Capability for retention of data and information

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

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

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

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

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

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

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

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

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

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

Response

When one discusses the post trip review as addressed in Generic Letter 83-28, Position 1.2, it encompasses more than just the classic Post Trip Review on the plant process computer. At Crystal River Unit 3, there are actually three systems which each provide data for analyzing plant transients, trips and restarts. These are: 1) the Post Trip Review on the plant process computer, 2) the annunciator events recorder system, and 3) the RECALL system.

The following paragraphs treat these three systems individually and provide a brief description and specify system design capabilities.

Plant Computer Post Trip Review

The plant process computer at CR-3 is a Modular Computer Systems unit, dual CPU system which was installed during the Refuel III outage and is powered from a vital buss. The current version of the Post Trip Review (PTR) implemented on the computer looks at forty (40) analog points. The points selected were defined by FPC as instrumental in the analysis of plant trips. The plant computer monitors the forty computer points on a fifteen (15) second scan frequency and stores them on a circulating disc file. The plant computer PTR software interprets a plant trip as: 1) opening both generator output breakers, or 2) opening either of two control rod drive mechanism's breakers. When a plant trip occurs, an additional fifteen minutes of plant data is collected on a fifteen second scan frequency and stored on disc. A message is provided to the operator that the PTR has been assembled. It will remain stored on non-volatile disc memory until the operator demands a hard copy printout.

Figure 1 below summarizes the PTR data collection scheme. See Attachment 2 of our 11/4/83 response for a sample PTR output showing format, and point identification. R1

15 minutes before trip	15 minutes after trip
15 sec scan rate	15 sec scan rate

Plant
Trip

Figure 1

Annunciator Events Recorder System

Crystal River Unit 3 has a Rochester Instrument System RA 800
events recorder system. The system is powered by a vital buss;
with the exception of the events printer which is powered by a R1
non-Class IE source. The system is designed to scan 2100 contact
inputs every 4.2 milliseconds, dedicating 2 microseconds to each
event point. Status of these contacts is provided on a hard copy
for permanent record. See Attachment 3 of our 11/4/83 response for R1
a brief sample.

RECALL

The RECALL system, installed during Refuel IV to meet NUREG-0696 requirements, is a data acquisition system with data storage, delog and real time functions. The data handler, input buffer and tape recorder storage hardware are located in the Emergency Feedwater Initiation and Control (EFIC) room while work stations for examining data are located in the control room and technical support center. The system is powered by a vital buss.

The RECALL system design capacity is 160 analog and 64 digital
points. Of this possible 224 points, approximately 206 are
currently defined and utilized. In addition, the 206 points
defined for the system have been assembled into 32 groups of
interrelated system parameters for operator convenience in
analyzing plant transients. Attachment 4 of our 11/4/83 response R1
provides a listing of these groups.

All system points and group assignments were defined by FPC as optimum for analyzing plant transients.

The RECALL system provides excellent data storage facilities. Each point is logged every second on a magnetic tape system. Capacity is four hours per cassette for four successive cassettes without operator intervention. At the end of this time, an operator must either remove any tapes that contain data that is to be saved or the system will automatically rewind and re-record on each of the four cassettes in succession.

The control room work station includes graphic display and hard copy facilities for parameter versus time information. It also has provisions for up to twelve strip chart channels.

Other Data and Information Sources

Strip Charts - provides hard copy of parameter trends.

RPS Cabinets - provides visual indication of which channels tripped.

Plant Computer Alarm Printout - provides hard copy of alarms received by the plant computer.

Schedule for Planned Changes

The current data and information capability is considered to be adequate to obtain an accurate understanding of plant transients at CR-3. Therefore, FPC is not actively pursuing changes to the existing capabilities.

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 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 back-up, 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.

Response

The B&W plant design does not incorporate a specific system which is designated as the "Reactor Trip System." Rather, specific portions of several systems are considered to comprise the reactor trip system. The following definition was applied to those systems to determine the specific components considered to comprise the reactor trip system: "Those components which must perform an active function to trip the reactor or whose failure would preclude a reactor trip, if demanded, are considered to be a portion of the reactor trip system."

FPC has verified that all safety related reactor trip system (RTS) components whose functioning is required to trip the reactor are properly classified on the "Safety Listing." Refer to the response to Position 2.2.1 for a description of the "Safety Listing" used to identify safety related components and the process used to control safety-related activities.

FPC has amended the "Safety Listing" to identify the non-safety related RTS components (e.g., silicon controlled rectifiers) that are to have safety related maintenance and test procedures applied per the requirements of Position 4.4. Although these RTS components are not safety related, their long operating history has proven them to be highly reliable. Upgrading of these RTS components to a safety grade status by design change or in procurement of safety related replacement parts is not deemed practicable as a means of improving reliability. As such, FPC does not intend to implement any design changes or changes in existing procurement practices. FPC does agree that the existing high RTS reliability can be maintained and possibly improved by expanding the scope of the safety related testing already being performed and by assuring that physical maintenance on these components is conducted using safety related procedures. R1

Refer to the response to Position 2.2.2 for a discussion of the current practice for handling vendor information and services. R1

B&W is the supplier of the reactor trip system components. As such, B&W has been tasked to identify the most current information for this equipment in order to verify that it has been received. Once it has been verified that the RTS information is current and complete, FPC will review the information to ensure that it is appropriately referenced or incorporated in plant procedures. To ensure that proper controls are implemented during this review, a program will be established to ensure each review will be consistent, no RTS components are omitted, and all RTS related procedures are reviewed. The review of the RTS procedures will be completed within 6 months of verification of existing vendor information or receipt of validated vendor information, but not later than December 31, 1984. If any changes to existing plant procedures or any new procedures are identified as a result of this review, they will be in place by March 1, 1985.

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 their programs for safety-related equipment classification and vendor interface as described below:

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

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

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

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

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 Position 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 non-nuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Response

1. FPC has in place and is maintaining programs to ensure that all components of safety-related systems necessary for accomplishing required safety functions are identified. The following items address the programmatic elements. R1

    1. FPC uses the "Safety Listing" to determine if structures, systems, or components are safety related or non-safety related. New structures, systems, or components are evaluated by Nuclear Engineering per Safety Related Engineering Procedure No. 1 (SREP-1), "Safety Identification Design Input Requirements," utilizing the following criteria as documented on the form "Safety Classification Review." See Attachment 5 of our 11/4/83 response. R1

        a. Does the item/service assure the integrity of the reactor coolant system boundary (i.e., "Pressure-retaining" as defined in ASME Boiler and Pressure Vessel Code)?

        b. Does the item/service assure the capability to shut down the reactor and to maintain it in a safe shutdown condition?

c. Does the item/service assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10CFR100.11?

All answers must be "No" for an item to be considered non-safety related.

The results of these reviews are incorporated into the "Safety Listing."

Individual piece parts of safety related structures, systems, or components are reviewed by Nuclear Engineering during the procurement cycle per the Nuclear Procurement and Storage Manual to determine if the piece part is safety related or non-safety related using the form "Classification of Items & Services." See Attachment 6 of our 11/4/83 response. An index which records the results of these reviews is maintained in the "Safety Listing." The criteria used in this determination is: R1

a. Does the product/service assure the integrity of the reactor coolant system boundary (i.e., "Pressure-retaining" as defined in ASME Boiler and Pressure Vessel Code)?

b. Does the product/service assure the capability to shut down the reactor and to maintain it in a safe shutdown condition?

c. Does the product/service assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10CFR100.11?

All answers must be "No" for a piece part to be considered non-safety related.

2. FPC uses a controlled "Safety Listing" to identify safety-related components. The original "Safety Listing" was developed by Gilbert Associates, Inc. (GAI), Reading, PA under contract to FPC. GAI was the A/E for the design of CR-3. The "Safety Listing" was subjected to independent design verification by GAI per their Procedures Manual and underwent a review and comment cycle by each FPC engineering discipline (electrical, I&C, mechanical, and structural), by plant staff personnel, and by the Quality Programs Department. Upon resolution of comments, the "Safety Listing" was issued under approval signature of each responsible discipline engineer and the Manager, Nuclear Engineering and Manager, Production Engineering. GAI used the following criteria in the preparation of the original "Safety Listing." A safety-related structure, system or component is:

One whose satisfactory performance is required:

a. To prevent accidents that could cause undue risk to the health and safety of the public; or

b. To mitigate the consequences of such accidents should they occur; or

c. To support and maintain the safe shutdown of the plant; or

d. A reactor coolant system pressure boundary.

NOTE: The criteria for a and b was 10CFR100.11.

Revisions to the "Safety Listing" are developed by an FPC design engineer and reviewed by all Nuclear Engineering supervisors, the Manager, Nuclear Engineering, the Nuclear Plant Manager, and the Quality Programs Department. Upon resolution of comments, the revision is subjected to independent design verification and issued under approval signature of the appropriate discipline supervisor and the Manager, Nuclear Engineering. The revision process is controlled by SREP-1.

The following Regulation, Regulatory Guide, National Standards, and references are consulted and used as applicable to CR-3:

1. 10 CFR Part 50
2. Regulatory Guide 1.26
3. ANSI N18.2a - 1975
4. ANSI N271 - 1976
5. IEEE 308
6. CR-3 FSAR and Facility Operating License including the Technical Specifications

The "Safety Listing" consists of two volumes. Volume 1 has seven sections as follows:

1. Introduction
2. Electrical Section
3. HVAC Section
4. I&C Section
5. Mechanical Section
6. Structural Section
7. Consumables Section

Volume 2 has two sections as follows:

1. Tab A - Copies of completed Safety Classification Review Forms until incorporation into Volume 1.

2. Tab B - Copies of the index of the Classification of Items Forms which record the results of Nuclear Engineering review of individual piece parts.

3. Plant personnel use the "Safety Listing" to determine if an activity is safety-related during the activity planning stage. Plant Operating Quality Assurance Manual (POQAM) Compliance Procedure (CP)-113, "Procedure for Handling and Controlling Work Requests and Work Packages," requires all Work Requests (WRs) identify that the activity involves safety-related or non-safety related components. This review is performed by the plant planners and documented on the WR by checking the appropriate box. CP-113 provides specific instructions to the planners to use the "Safety Listing" to make this determination. Since all maintenance at CR-3 is done by WRs, this review ensures that maintenance, spare parts and post-maintenance testing are correctly addressed. Surveillance testing is performed in accordance with CR-3 Technical Specification requirements and NRC commitments in accordance with approved procedures.

   In addition, CP-113 requires a review of the WR prior to its being worked by the Shop Supervisor and the Shift Supervisor. This review provides adequate, independent review of the WR as to its safety-related or non-safety related status.

   All WRs that are indicated as safety-related are stamped "If parts are required, and quality parts are not available, contact QC." This statement assures that only properly qualified replacement parts are used on safety-related WRs.

   POQAM Administrative Instruction (AI)-401, "Origination and Implementation of New Procedures," and AI-402, "Making and Implementing Revisions to POQAM Procedures," have been revised to require a statement in applicable new POQAM procedures to address whether the structure, system, or component to which it applies is safety-related or non-safety related. In addition, applicable existing POQAM procedures have been reviewed and the appropriate statement added to identify whether the structure, system, or component to which they apply are safety-related or non-safety related. R1

   The "Safety Listing" is reviewed by design personnel during the modification process to determine if the modificiation affects safety-related structures, systems, or components. The results of this review are documented on a "Design Data Sheet" Form which is included in the Modification Package. This process is controlled by SREP-1 and SREP-6, "Preparation and Control of a Modification Approval Record (MAR)." New structures, systems, or components being added to CR-3 are evaluated as discussed in the response to Position 2.2.1.1.

4. The management controls utilized to verify that the procedures for preparation, validation and routine utilization of the "Safety Listing" have been followed are:

   a. Nuclear engineering supervisory approval of revisions to the "Safety Listing," Safety Classification Review Forms, Classification of Items & Services Forms, Procurement Packages, and Design Change Packages (MARs).

   b. Nuclear plant supervisory approval of all Work Requests.

   c. Manager, Nuclear Engineering approval of SREP procedure changes and revisions to the "Safety Listing."

   d. Audits by Quality Programs Department of maintenance and design control activities.

5. Florida Power Corporation's demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components can be separated into two parts.

   First, the method by which replacement parts are ordered for equipment that has been deemed qualified for the environment in which it is installed.

   Second, the method by which new equipment is ordered.

   In both cases, the procedures that govern all material transactions at Crystal River Unit 3 are contained in the Nuclear Procurement and Storage Manual. The environmental parameters for safety-related electrical equipment are contained in Florida Power Corporation specification SP-5095, "Environmental & Seismic Qualification Guide," in the form of zone maps. The zones within the plant are broken into two basic categories, i.e., harsh and mild.

   For replacement parts of qualified equipment, Section 6.3 of the Nuclear Procurement and Storage Manual permits the use of the "Catalog" method of procurement which represents a safety-related commercial grade method. The replacement part will be ordered from the original manufacturer with either a requirement for a Configuration Certificate or a Certificate of Conformance that the part ordered is the same as the part tested in the original equipment including a reference qualification report and date. The Configuration Certificate would also reference a report number and date. An example of such a procurement package was included as Attachment 7 in our 11/4/83 response. R1

For the new equipment that requires qualification testing, Section 6.1 of the Nuclear Procurement and Storage Manual requires the use of the "Specification" method of procurement which represents a nuclear grade method invoking 10CFR50 Appendix B and 10CFR21. The technical requirements of the purchase would include the environmental parameters (both design basis events and normal) in which the equipment must operate using the SP-5095 zone maps. Engineering software in the form of an Equipment Qualification Report is required to be submitted for acceptance prior to shipment. The report is reviewed for compliance with the requirements of NUREG-0588 Category 1 for harsh environments. An example of this material transaction was included as Attachment 8 in our 11/4/83 response. R1

6. The equipment classification program for safety-related components is as described above. FPC has two classifications, safety related and non-safety related. With respect to the equipment classification program for structures, systems and components important to safety, FPC is participating in the Utility Safety Classification Group and is 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, systems and components important to safety constitute a broader class than the safety related set. We nevertheless believe that non-safety related plant structures, systems and components have been designed, and are maintained, in a manner commensurate with their importance to the safe and reliable operation of the plant. R1

2. FPC has actively participated in the Institute of Nuclear Power Operations (INPO) sponsored Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Position 2.2.2. This NUTAC was formed for the specific purpose of defining an appropriate vendor interface program. The NUTAC's goal was to enhance vendor information exchange and evaluation among utilities. The NUTAC developed the Vendor Equipment Technical Information Program (VETIP) to combine existing information exchange programs with a coordinated evaluation program within each utility. This concept was unanimously endorsed by the NUTAC and was presented in the NUTAC's final report dated March 23, 1984. FPC considers the VETIP to be responsive to Generic Letter Positions 2.1 and 2.2.2. FPC has completed its review of our existing programs and procedures against the recommendations of the NUTAC Final Report and has modified our current practices to support the VETIP. R1

In order to implement this program, FPC will review the available information for safety related components to ensure it is appropriately referenced or incorporated in plant procedures. FPC will also review plant procedures for safety related components where vendor information is not available to ensure sufficient attention is paid to maintenance, replacement, and repair. To ensure that proper controls are implemented during these reviews, a program will be established to ensure each review will be consistent, include all safety related components, and include all safety related procedures. The review of the safety related procedures will commence once it is known that vendor information is current and complete to the best of our knowledge and will be completed as part of the 24 month procedure review cycle, but not later than December 31, 1985. If any changes to existing plant procedures or any new procedures are identified as a result of this review, they will be in place by March 3, 1986. |R1

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

Position

The following actions are applicable to post-maintenance testing:

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

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

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

Response

1. At this time, FPC is in compliance with Technical Specifications. Post-maintenance testing is performed on safety-related reactor trip system components once maintenance is complete. Compliance Procedure CP-113, "Procedure for Handling and Controlling Work Requests and Work Packages," delineates the FPC requirements for performing post-maintenance testing. Additionally, existing maintenance procedures have a requirement to address specific post-maintenance testing. At this time, the completeness of this section is considered adequate.

   FPC will review each procedure affecting maintenance to safety-related components in the reactor trip system to ensure the post-maintenance test section adequately demonstrates that the equipment is capable of performing its safety function before being returned to service. This review will coincide with the review being performed per our response to Position 2.1.

   The review of reactor trip system information will be completed within 6 months of verification of existing vendor information or receipt of validated vendor information, but not later than December 31, 1984. If any changes to existing plant procedures or any new procedures are identified as a result of this review, they will be in place by March 1, 1985.

2. Procedures were written initially using vendor information and engineering recommendations as applicable. While no formal program existed to document our effort, vendor and engineering recommendations have been incorporated in our procedures to ensure adequate test guidance.

   FPC will review the test and maintenance procedures and Technical Specifications for safety-related components in the reactor trip system to ensure that the test guidance reflects appropriate vendor information and engineering recommendations. This review will document our efforts to include vendor information and engineering recommendations in the test and maintenance procedures and Technical Specifications, where required. This review will coincide with the review being performed per our response to Position 2.1. The review of the RTS information will be completed within 6 months of verification of existing vendor information or receipt of validated vendor information, but not later than December 31, 1984. If any changes to existing plant procedures or any new procedures are identified as a result of this review, they will be in place by March 1, 1985.

3. FPC has not identified any post-maintenance test requirements which we can demonstrate to degrade safety at this time. However, the ongoing reviews identified above or others could identify such demonstrated degradations in safety. If and when such are identified, FPC will submit appropriate Technical Specification Change Requests to the Commission.

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

Position

The following actions are applicable to post-maintenance testing:

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

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

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

Response

1. At this time, FPC is in compliance with Technical Specifications. Post-maintenance testing is performed on safety-related components once maintenance is complete. Compliance Procedure CP-113, "Procedure for Handling and Controlling Work Requests and Work Packages," delineates the FPC requirements for performing post-maintenance testing. Additionally, existing maintenance procedures have a requirement to address specific post-maintenance testing. At this time, the completeness of this section is considered adequate.

   FPC will review each procedure affecting maintenance to safety-related components to ensure the post-maintenance test section adequately demonstrates that the equipment is capable of performing its safety function before being returned to service. This review will coincide with the review being performed per our response to Position 2.2.2. This review will be completed as part of the 24 month procedure review cycle, but not later than December 31, 1985. If any changes to existing plant procedures or any new procedures are identified as a result of this review, they will be in place by March 3, 1986. R1

2. Procedures were written initially using vendor information and engineering recommendations as applicable. While no formal

program existed to document our effort, vendor and engineering recommendations have been incorporated in our procedures to ensure adequate test guidance.

FPC will review the test and maintenance procedures and Technical Specifications to ensure that the test guidance reflects appropriate vendor information and engineering recommendations. This review will document our efforts to include vendor information and engineering recommendations in the test and maintenance procedures and Technical Specifications, where required. This review will coincide with the review being performed per our response to Position 2.2.2. The review will be completed as part of the 24 month procedure review cycle, but not later than December 31, 1985. If any changes to existing plant procedures or any new procedures are identified as a result of this review, they will be in place by March 3, 1986. R1

3. FPC has not identified any post-maintenance test requirements which we can demonstrate to degrade safety at this time. However, the ongoing reviews identified above or others could identify such demonstrated degradations in safety. If and when such are identified, FPC will submit appropriate Technical Specification Change Requests to the Commission.

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 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

Our 11/4/83 response indicated that there have been no vendor-recommended field modifications from either B&W or General Electric to the General Electric Model AK-2-25 reactor trip breakers. This fact was documented in writing by General Electric. See Attachment 9 of our 11/4/83 response. FPC has not initiated any modifications.

Subsequent to that response, it was identified that certain RTBs manufactured during the period November 1976 through March 1977 were manufactured with a defect. It has been verified that no FPC RTBs were effected.

During a meeting with General Electric (GE) in February 1984, the B&W Owners Group (BWOG) first learned of new information and recommendations from GE with regard to GE AK reactor trip breakers (RTBs). GE suggested that use of WD 40 to rejuvenate the Lubriko lubrication in the trip shaft bearings was having an accelerated effect on the lubrication aging (hardening) process. GE further indicated that they were conducting extensive testing of new lubricants for use as a replacement to the Lubriko and that preliminary results indicated Mobil 28 was an excellent replacement. GE recommended that WD 40 only be used for rejuvenation when trip shaft torque approaches unacceptable limits and that consideration be given to replacing the bearings with bearings lubricated with Mobil 28 lubricant. Documentation was provided by GE Service Advice 9.20. R1

Although not specifically recommended by the GE Service Advisory or Generic Letter 83-28, the BWOG began development of a program to evaluate long term improvements in breaker performance. The RTB Long Term Improvement Program is intended to evaluate a broad scope of alternatives to improve RTB reliability.

Two categories of fixes are being evaluated:

Modification of existing GE AK breakers; and

Replacement of GE AK breakers with a different device.

After the list of alternatives was developed, evaluation criteria were established to be used in the selection process. The evaluation is being conducted using the Kepner-Tregoe Decision Analysis method. The first round of evaluation has now been completed which indicates that alternatives for modification of the existing GE AK RTBs are superior alternatives to those which involve replacement of the RTBs with a new/different device. This is strongly based on the extremely positive results of GE's lubrication testing demonstrating the stable nature of Mobil 28 lubricant, GE's agreement to continue making parts available for the AK RTBs past the year 2000, and the increased trip margin from the Undervoltage device that results with the Mobil 28 lubricated bearings. As such, further evaluation is concentrating on only "modification" alternatives. A final recommended fix is expected to be available in September for presentation to and approval by the entire Owners Group. The BWOG will be requesting a meeting with the NRC to discuss the final results of the evaluation program. As it now seems clear that the final fix will use the existing GE AK RTBs, FPC will continue to replace the bearings lubricated with Lubriko in a timely manner with bearings lubricated with Mobil 28. R1

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:

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

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

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

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

Response

1. As stated in our 11/4/83 response, FPC has in place a procedure for periodic maintenance that includes the General Electric Company recommended maintenance items (Service Advice 9.3 and 9.3S). A copy of Preventative Maintenance Procedure PM-118, "AC and DC Breakers-Control Rod Drive System," Rev. 7, was included as Attachment 10 in our 11/4/83 response.

   After the initial incorporation of the GE guidance, the B&W Owners Group collectively reviewed the incorporation of the GE guidance in each utility's procedures to identify any differences or discrepancies. Those items identified from that review have been incorporated into PM-118. R1

   Screening and operability criteria have been incorporated into PM-118 to be used to determine the need for maintenance and/or operability of the RTB. The new GE recommendation (Service Advice 9.20) for use of WD-40 only on an "as needed basis" has also been incorporated into PM-118.

2,3,&4. FPC is participating in the B&W Owners Group Reactor Trip Breaker (RTB) Reliability Monitoring Program. The RTB Reliability Monitoring Program will compile and analyze maintenance and surveillance data for the General Electric AK-2 RTBs. The program has the following objectives:

   1. Accumulate a substantial amount of breaker test and maintenance data for the purpose of defining current breaker reliability and providing a basis for evaluating future actions taken to improve breaker performance.

2. Provide a means for users of AK-2 breakers to track the performance of AK-2 breakers relative to their own application and that of other users.

3. Provide a means to collect and disseminate information among users relative to breaker failures and corrective actions.

Data will be collected during the performance of surveillance and maintenance activities. Data from all AC and DC RTBs will be obtained (six breakers for each Oconee-series reactor and four breakers for each Davis-Besse-series reactor).

The data to be collected includes undervoltage device response times, as-found and as-left trip bar torque values, and as-found and as-left undervoltage device pickup and dropout voltage setpoints. The data will be sent by each utility to B&W who will act as the compiler. B&W will compile the data and issue periodic reports to the utilities to enable each utility to compare its breaker performance with the norm defined by the data base. Consideration will be given to the type of performance monitoring that may be necessary to demonstrate the acceptability of the long term RTB modification discussed in the response to Position 4.1. The data collection and compilation program is expected to be run for two years after initiation. R1

The number of RTB tests and failures is currently being collected by the B&W Owners Group.

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 attachments. The shunt trip attachment shall be considered safety-related (Class IE).

Response

FPC will provide automatic actuation of the reactor trip breakers (RTBs) shunt trip coils to provide a backup to the undervoltage coils for tripping the RTBs. The shunt trip actuation design for CR-3 is based upon the Arkansas Power and Light (AP&L) design concept. The AP&L concept utilizes additional undervoltage relays of the solid state type. The solid state relays monitor power from the Reactor Protection System and are actuated by undervoltage in the same manner as the existing AC & DC RTB undervoltage coils. Digital outputs from the solid state relays are used to apply 125 volt DC power to the AC & DC RTB shunt trip coils. Testability is provided for in the design to allow for independent verification of RTB trip by the undervoltage coil and by the shunt trip coil. Independent verification is accomplished by use of manual test switches. Loss of 125 volt DC power at the shunt trip coil is alarmed on a control room annunciator. R1

The AP&L design was presented to the NRC and identified as a generic approach for B&W utilities. The NRC reviewed this design and issued a Safety Evaluation Report (SER) on September 12, 1983. Enclosure 1 to the SER lists eight items to be addressed on a plant specific basis.

FPC submitted preliminary design information for the CR-3 response to the eight SER items on May 30, 1984. On June 19, 1984, FPC committed to supply final design information by July 31, 1984. Attachment A to this letter provides the final design information. FPC requests prompt review and approval of our design in order to allow us to proceed with installation during the next outage of sufficient duration.

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

Position

Licensees and applicants with B&W reactors shall apply safety-related maintenance and test procedures to the diverse reactor trip feature provided by interrupting power to control rods through the silicon controlled rectifiers.

This action shall not be interpreted to require hardware changes or additional environmental or seismic qualification of these components.

Response

Silicon controlled rectifier maintenance and testing requirements have been incorporated into Surveillance Procedure SP-110, "Reactor Protective Systems Functional Test," and Preventative Maintenance Procedure PM-118, "AC and DC Breakers - Control Rod Drive System."

A Technical Specification Change Request was submitted on January 16, 1984 to include the silicon controlled rectifiers in the surveillance requirements. R1

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.

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

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

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

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

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

Response

1. FPC performs on-line testing of the undervoltage trip feature and the circuitry used for power interruption with the silicon controlled rectifiers monthly per SP-110, "Reactor Protection System Functional Testing." Circuitry to perform on-line testing of the shunt trip feature is provided for in the design. See response to Position 4.3. |R1

2. All components of the reactor trip system, as installed at Crystal River Unit 3, are currently testable with the plant on-line with the exception of the reactor trip switch on the main control board. However, this switch is provided only for operator initiation of a reactor trip and is completely manual in operation. As such, it does not fall under the guidelines listed in NUREG-0800, Section 7.2 - Reactor Trip System, for those devices which should have on-line testing capability. Therefore, no modifications are required to the RTS as presently installed at CR-3 to provide on-line testability.

   The shunt trip feature is provided with circuitry for testability on-line. See response to Position 4.3. |R1

3. FPC is particpating in the B&W Owners Group sponsored activity to demonstrate by analysis that the one-month test interval, currently used for on-line testing of the RTS, is consistent with high RTS availability in consideration of the factors cited in the above position. This analysis is expected to be completed by December 1984. At the conclusion of this effort, if FPC has sufficient technical basis to revise test intervals, FPC will submit such Technical Specification Change Requests.

ATTACHMENT A

INFORMATION REQUIRED ON A PLANT SPECIFIC BASIS FOR REVIEW AND STAFF APPROVAL OF MODIFICATIONS TO PROVIDE AUTOMATIC ACTUATION OF REACTOR TRIP BREAKER SHUNT TRIP ATTACHMENTS

Item 1:

A statement confirming that the UV sensor (high speed undervoltage relay) Model ITE-27H-211R, is environmentally and seismically qualified for its service conditions.

Response 1:

The undervoltage sensor, ITE-27H-211RO175, has been environmentally and seismically qualified by Brown Boveri Electric Incorporated to IEEE 323-1974, IEEE 501-1978, and IEEE 344-1975. The sensor is qualified for -20°C to +55°C, 0 to 90% relative humidity (no condensation) and 6g ZPA. These qualifications envelop the service conditions at the sensor mounting location. The ITE-27H-211RO175 is therefore qualified for use in the control rod drive trip breaker shunt trip application at Crystal River 3.

Item 2:

A statement confirming that all other additional components involved in the shunt trip circuits are environmentally and seismically qualified for their service conditions.

Response 2:

The additional equipment used in the control rod drive trip breaker shunt trip application at Crystal River 3 is:

Potter Brumfield Rotary Relay	-	MDR 138-8
Electro Switch	-	Series 20P
Buchanan Terminal Blocks	-	NQB 104 & NQB 106

Potter Brumfield Rotary Relay MDR 138-8 will be qualified by NU-Therm International Incorporated to IEEE 323-1974, IEEE 501-1978 and IEEE 344-1975. The environmental and seismic test conditions will envelop the service conditions at the relay mounting location at Crystal River 3.

Electro Switch Series 20P has been qualified by Electro Switch Corporation to IEEE 323-1974 and IEEE 344-1975. The Electro Switch Series 20P is qualified for 80°C (120 hours), 95% relative humidity (96 hours) and 5g ZPA. These qualifications envelop the service conditions at the switch mounting location. The Electro Switch Series 20P is therefore qualified for use in the control rod drive trip breaker shunt trip application at Crystal River 3.

Buchanan terminal blocks NQB 104 and NQB 106 have been qualified by Amerace Corporation to IEEE 323-1974 and IEEE 344-1975. The Buchanan terminal blocks NQB 104 and NQB 106 are qualified for temperatures up to 150°C and 5g ZPA. These qualifications envelop the service conditions at the mounting location. The Buchanan NQB 104 and NQB 106 terminal blocks are therefore qualified for use in the control rod drive trip breaker shunt trip application at Crystal River 3.

Item 3:

A statement confirming that the shunt trip attachment is or will be environmentally and seismically qualified for its service conditions.

Response 3:

The shunt trip attachment is environmentally and seismically qualified for its service conditions.

Item 4:

Identify the classification (safety related or not) and separation (train or channel identification) for the reactor trip shunt and UV trip circuits, power supplies, and any interface isolation devices.

Response 4:

The control rod drive trip breaker undervoltage (UV) and shunt trip circuits are safety related. Safety related power sources are used to power the undervoltage and shunt trip circuits of the control rod drive trip breakers. Separation of power divisions has been maintained by use of conduit, barriers, and/or separation distances of six inches or more. Points where less than six inches separation distance occur are addressed in Response 5.

Power Division/Channel Assignments are as follows:

Control Rod Drive Trip Breaker	Undervoltage Coil Channel	Shut Trip Coil Channel
AC (Unit 10)	Vital Bus 3A (RPS CH A)*	DC Bus 3A (DPDP 5A)**
AC (Unit 11)	Vital Bus 3B (RPS CH B)	DC Bus 3B (DPDP 5B)
DC (Bkrs 1&2)	Vital Bus 3C (RPS CH C)	DC Bus 3B (DPDP 5B)
DC (Bkrs 3&4)	Vital Bus 3D (RPS CH D)	DC Bus 3A (DPDP 5A)

* RPS - Reactor Protection System
** DPDP - DC System Distribution Panel

The above buses are shown in the Crystal River 3 FSAR Figure 8-8.

The interface between the safety related DC power supply and the non-safety related plant annunciator, loss of DC shunt trip power alarm, is accomplished through the coil to contact isolation of the Potter Brumfield MDR 138-8 relay.

The control rod drive AC trip breakers are equipped with a source interrupt device. The source interrupt device actuates the AC trip breaker shunt trip coil upon overvoltage or undervoltage of the supply bus. This function is to protect the holding coils for the control rod drives and is not considered safety-related.

The non-safety related source interrupt is isolated from the safety related shunt trip circuit through the coil to contact isolation of the Potter Brumfield MDR 138-8 relay.

Item 5:

If the wiring to the UV sensor involves different separation groups (train or channel) identify the minimum separation (distance) between wiring of the different groups. Provide an analysis of the consequences of short circuits between wiring in different separation groups to confirm that the consequences do not adversely impact redundant safety related systems.

Response 5:

Wiring to the UV sensor for the control rod drive AC trip breakers does not involve different power divisions and, therefore, does not pose a concern. Wiring to the UV sensor for the control rod drive DC trip breakers does involve different power divisions - 120 VAC of one division and 125 VDC of the opposite division. This same case exists for the UV and shunt trip attachments on the control rod drive DC trip breakers. Regulatory Guide 1.75 which generally endorses IEEE 384-1974 was consulted for guidance in this area.

Section 5.6.2 of IEEE 384-1974 states that the minimum separation distance can be established by analysis of the proposed installation, and that 6 inches shall apply where analysis has not been performed. With two exceptions, the separation distance utilized is 6 inches. The two exceptions are (1) at the terminals of the UV sensor, and (2) at the shunt and UV trip attachments within the control rod drive DC trip breakers. For this reason, the consequences of a short circuit between the 120 VAC and 125 VDC have been analyzed. The results of this analysis are provided as follows:

1. Because of the existence of a constant voltage transformer in the inverter, the 125 VDC cannot propagate back through the inverter to the other DC bus.

2. The shorting together of one leg from each circuit would theoretically have no effect except to introduce a reference point to the ungrounded DC system.

3. The shorting together of both legs of 120 VAC to both legs of 125 VDC would result in the blowing of one or both branch fuses. The main breakers/fuses are not jeopardized due to proper coordination over the range of available fault current.

4. Under the worst case in Item 3 above, the system will perform its required safety function. The 120 VAC and 125 VDC circuits involved do not supply redundant components from a system viewpoint.

Based on the above analysis, this system is considered to meet the requirements of a safety-related design in accordance with the standards referenced in the FSAR.

Item 6:

Provide an outline of the test procedures to independently verify the operability of the shunt and UV trip circuits and components. Identify the sequence of actions to be performed. Address your intent regarding periodic surveillance to confirm the operability of the power failure alarms.

Response 6:

The loss of DC shunt trip power alarms will be tested each time the trip circuits are tested.

An outline of the proposed test procedure sequence to independently verify the operability of the shunt and UV trip circuits is given in Attachment B.

Item 7:

Provide a draft of any proposed technical specification changes as a result of this modification.

Response 7:

Existing CR-3 Technical Specifications governing operability and surveillance of the Reactor Protection System and control rod drive trip breakers envelop operability and surveillance requirements for the shunt trip. As such, no changes to the existing Technical Specifications are deemed necessary. Appropriate plant procedures will be changed to reflect installation of the shunt trip modification.

Item 8:

Provide the electrical schematics for the shunt and UV trip circuits.

Response 8:

The electrical schematics for the shunt and UV trip circuits are Attachment C.

ATTACHMENT B

Control Rod Drive Trip Breaker Test Outline

The following test outline requires use of all four RPS Channels and associated control rod drive trip breakers. Test steps should be performed such that breaker cycle times are not exceeded.

1.0 Initial Conditions

1.1 Notify Shift Supervisor before starting test.
1.2 Verify that each of the following breakers are closed.
   a. Main breaker feeding each control rod drive AC trip breaker.
   b. Circuit breaker for 125VDC control power feeding each control rod drive trip breaker.
1.3 Verify that the UV sensor signal (120VAC from RPS) indicator lights and 125VDC shunt trip power indicator lights are "on" at each control rod drive trip breaker test switch.

NOTE: The following steps require RPS channel trips from the RPS cabinets located in the control room. Establish communications as required. Local test switches are spring return and must be held in the required test position.

2.0 Undervoltage trip circuit testing.
(Typical for two CRD AC trip breakers and two sets of two CRD DC trip breakers).

2.1 Place local test switch in the "UV coil test" position.
2.2 Verify that the 125VDC shunt trip power indicator light is "off".
2.3 Verify that the "Loss of CRD BKR shunt trip PWR" alarm has actuated in the control room.
2.4 In the respective RPS channel, at the reactor trip module place two trip test switches to the "trip" position.
2.5 Verify breaker tripped. (NOTE: for CRD DC breaker, verify both breakers tripped.)
2.6 Verify that the UV Sensor signal (120VAC from RPS) indicator light is "off".
2.7 Release local test switch.
2.8 Verify that the 125 VDC shunt trip power indicator light is "on".
2.9 Verify that the "Loss of CRD BKR shunt trip PWR" alarm has cleared in the control room.
2.10 In the respective RPS channel, at the reactor trip module return the two trip test switches to the "normal" position.
2.11 Reset the reactor trip module.
2.12 Verify that the UV Sensor signal (120VAC from RPS) indicator light is "on".
2.13 Close the control rod drive trip breaker.

3.0 Source Interrupt Test
(Control rod drive AC trip breakers only).

3.1 Momentarily push source interrupt trip test button.
3.2 Verify that CRD AC breaker tripped.
3.3 Close the control rod drive trip breaker.

4.0 Shunt trip circuit testing
(Typical for two CRD AC trip breakers and two sets of two CRD DC trip breakers.)

4.1 Place local test switch in the "shunt coil test" position.
4.2 Verify breaker tripped. (NOTE: for CRD DC breaker, verify both breakers tripped.)
4.3 Verify that the UV Sensor signal (120VAC from RPS) indicator light is "off".
4.4 Release local test switch.
4.5 Verify that the UV sensor signal (120VAC from RPS) indicator light is "on".
4.6 Close the control rod drive trip breaker.

5.0 Test CRD breaker performance on loss of 125VDC shunt trip power.

5.1 Open the circuit breaker supplying 125VDC power to the CRD breaker shunt trip circuit.
5.2 Verify that the control rod drive trip breaker did not trip.
5.3 Verify that the 125VDC shunt trip power indicator light is "off" for the control rod drive trip breaker being tested.
5.4 Verify that the "Loss of CRD BKR shunt trip PWR" alarm has actuated in the control room.
5.5 Close the circuit breaker supplying 125VDC power to the CRD breaker.
5.6 Verify that the 125VDC shunt trip power indicator light is "on".
5.7 Verify that the "Loss of CRD BKR shunt trip PWR" alarm has cleared in the control room.
5.8 Verify that the control rod drive trip breaker did not trip.

6.0 Testing Completion

6.1 Notify Shift Supervisor that the testing is complete.

7.0 Test Frequency

7.1 Sections 1.0, 2.0, 4.0 and 6.0 are to be performed monthly and prior to plant startup if not previously performed within seven days.
7.2 Sections 3.0 and 5.0 are to be performed prior to plant startup if not previously performed within seven days.

ATTACHMENT C

CONTROL ROD DRIVE TRIP
BREAKER ELECTRICAL SCHEMATICS

1. Drawing No. 208-024C Rev. 0, Sheet DR-03; Elementary Diagram Primary Breaker Shunt Trip Unit 10 & 11.

2. Drawing No. 208-024C Rev. 0, Sheet DR-04; Elementary Diagram DC Breaker Shunt Trip Unit 1 & 2.

3. Drawing No. 208-024C Rev. 0, Sheet DR-05; Elementary Diagram DC Breaker Shunt Trip Unit 3 & 4.