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50-389 St. Lucie Plant, Unit 2, Florida Power & Light Co. 05000389

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SUBJECT: Forwards procedures to support util actions re Generic Ltr 83-28 & on generic implications of Salem ATWS events.

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FLORIDA POWER & LIGHT COMPANY

November 8, 1983
L-83-554

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

Dear Mr. Eisenhut:

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

Florida Power & Light has reviewed Generic Letter 83-28 (Required Actions Based on Generic Implications of Salem ATWS Events) and a response is attached.

Should you or your staff have any questions on the attached information, please contact us.

Very truly yours,

J. W. Williams, Jr.
Vice President
Nuclear Energy Department

JWW/PLP/js

Attachment

cc: Mr. James P. O'Reilly, Region II
Harold F. Reis, Esquire
PNS-LI-83-696

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PEOPLE... SERVING PEOPLE

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

The program for ensuring that unscheduled reactor shutdowns are analyzed and a determination is made that the plant can be restarted safely is described for St. Lucie Units 1 & 2 in Operating Procedure 0030119, attached. The following comments address the items in the staff's position:

1. NRC Position

   The criteria for determining the acceptability of restart.

   Response

   Step 19 of the procedure requires that the cause of the trip has been identified, that no abnormal conditions exist, and that the plant can be safely returned to power.

2. NRC Position

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

   Response

   As shown in the referenced procedure, the Nuclear Plant Supervisor and the Shift Technical Advisor are both required to sign the procedure and to conclude that the plant can safely be returned to power.

3. NRC Position

   The necessary qualifications and training for the responsible personnel.

   Response

   Plant Manager - Meets ANSI/ANS 3.1 1978 standards. He shall have acquired the experience and training normaly required for examination by the NRC for an SRO at St. Lucie, whether or not the exam is taken.

   Operations Superintendent - Meets ANSI/ANS 3.1 1978 standards.

   Operations Supervisor or Shift Supervisor - Meets ANSI/ANS 3.1 1978 standards. He shall hold a current SRO license on PSL 1 and 2.

   Assistant Nuclear Plant Supervisor (ANPS) - Meets ANSI/ANS 3.1 1978 standards. He shall hold a current SRO license on PSL 1 or 2.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

Shift Technical Advisor (STA) - He shall hold a bachelors degree or equivalent in a scientific or engineering discipline. He shall be trained in the response and analysis of the plant for transients and accidents. He shall be trained in the details of the design, function, arrangement, and operations of plant systems, including the capabilities of instrumentation and controls in the control room.

A Technical Specification change is being sought to allow STA function to be incorporated into ANPS position provided that the ANPS has a degree in engineering and meets the training requirements of the STA function.

4. NRC Position

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

Response

The plant information used to conduct the review and analysis is described in our response to Action 1.2.

5. NRC Position

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

Response

The methods and criteria for comparing the event information with known or expected plant behavior are provided in the procedure in steps (2) through (13) and step (18).

6. NRC Position

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.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

Response

The procedure states that if the cause of the plant trip cannot be determined by the NPS the unit will not be returned to power until the events associated with the trip are evaluated by the Facility Review Group. Step (17) requires that the Digital Data and sequence of events printouts and any other printout or chart used to determine the cause of the reactor trip or to show any abnormal condition during trip be attached to the checksheets, which shall be retained in the plant files.

7. NRC Position

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

The procedure is attached.

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

NRC Position

Licensees and applicants shall have or have planned a capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment.

Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Action 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

Response

This report describes that equipment used by plant personnel to determine proper safety equipment/system operation and assess causes for unscheduled reactor shutdowns for St. Lucie Units 1 and 2. In some cases the descriptions for both Units are combined when sufficient similarities exist.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

Sequence of Events Recorder (SER)

The main source of post trip information is the Sequence of Events Recorder (SER). Input to these devices is change-of-state indications. Many key plant inputs are monitored by the SER. These include safety related pump conditions (on/off), major essential and non-essential 4160V and 6.9 KV breaker status (open/closed/trip/reset), initiating events for reactor trip and engineered safeguards actuations (trip/reset). Attached is a listing of the parameters monitored by the Unit 1 SER (Attachment 1). Parameters monitored by the Unit 2 SER are almost identical.

Unit 1 utilizes a Fischer Porter Series 3000 system while Unit 2 utilizes a Rochester Instruments System model RA-800L.

All inputs are scanned once in 2 milliseconds by the Fischer Porter system (Unit 1) and once every 1 millisecond by the Rochester Instruments System (Unit 2). This means a maximum or worst case time discrimination between events would be 2 milliseconds and 1 millisecond for Unit 1 and 2, respectively. Both systems possess a "debounce" feature which requires a contact be maintained for 5 milliseconds or more before it is recognized for printing out.

Output for both SERs is hardcopy typed. The format for the output contains point identification number or input number, contact message status (i.e., on/off, open/close, stop/start, trip/reset, raise/lower and normal/abnormal), time of day to the millisecond with corresponding equipment description.

Both Unit 1 and Unit 2 SERs are powered from the Static Uninterruptable Power Supply (SUPS). Unit 1 SER is powered from vital bus 1 and Unit 2 SER is powered from vital bus 2A, both are non-Class IE. Both buses are station battery powered on a Loss of Off-Site Power and capable of being loaded on the emergency diesels.

All SER printouts are permanently retained in the site QC vault.

Analog Variables

Assessing the time history of analog variables is accomplished mostly by permanent plant chart recorders located in the control room. Approximately 100 recorders are continuously monitoring key parameters. Examples of parameters monitored are: emergency diesel generator watts, steam generator pressure and levels, pressurizer level, pressure and temperature, reactor coolant system loop temperatures, reactor power, HPSI, LPSI and containment spray flow and pressures, and plant process and area radiation levels. Fischer Porter, Leeds & Northorp, and Foxboro constitutes the majority of recorders at St. Lucie Units 1 and 2.

Typical chart scale is 4 inches with a time base of 1 inch per hour. This type of information provides excellent long-term data prior to and following an event.

Almost all strip charts are retained on-site for either system for 5 years or life of the plant. A few redundant or less frequently used recorder charts are not retained.

Safety Assessment System (SAS)

The Safety Assessment Systems designed for St. Lucie Units 1 and 2 are essentially identical. They monitor approximately 1000 parameters per unit and supply this information to the Technical Support Center on-site and the Emergency Operations Facility off-site, as well as the control room if requested.

This system is supplied by Technology for Energy (TEC) Corporation and is capable of multi-scan rates dependent upon parameter chosen. Scan rates can be programmed to vary from 1 second to 60 seconds. Also the system can retain two hours pre-event information once activated by operations personnel and continue recording for 10 hours after activation. The system design and performance requirements are in accordance with NUREG-0696. Power for the SAS is supplied via vital DC buses, Non-Class IE, and is capable of being loaded on an emergency diesel generator.

Hardware for SAS on Unit 1 is now installed with software checkout and generation in progress. SAS for Unit 1 should be functional shortly after return to power. St. Lucie Unit 2 SAS is scheduled for installation during its first refueling outage. Installation of the SAS will greatly increase the amount and quality of post trip information.

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

NRC Position

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

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

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 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

FPL has determined that at St. Lucie Units 1 and 2 components whose functioning is required to trip the reactor are included in systems which are treated as safety related for plant activities, such as maintenance, work orders and parts replacement. Future changes or modifications of these systems will be reviewed by FPL engineering to ensure that the correct safety classification is made. A description of the information handling systems for component classification is provided in our response to action 2.2.1.

In addition a continuing program exists to receive and review vendor information. The Availability Data Program is a system for reporting information that could affect the performance and reliability of operating plants utilizing Combustion Engineering (CE) Nuclear Steam Supply Systems. "INFOBULLETINS" are issued as part of this program with information and recommendations concerning CE systems. A return receipt is required for these bulletins. Upon receipt, these bulletins are entered into the FPL Operating Experience Feedback Program for review and implementation where applicable. These items are tracked as part of this system until completion.

FPL Quality Procedures 4.1 (Control of Requisitions and the Issuance of Purchase Orders to spare parts, replacement items and services) and 4.4 (Review of Procurement Documents for Items and Services Other than Spare Parts) provide a system to assure that the appropriate technical and quality requirements are placed upon suppliers who provide material, equipment and services for operating nuclear power plants. Safety-related Purchase Orders for services are issued only to suppliers whose quality Assurance Program and implementing procedures have been evaluated and approved by the FPL Quality Assurance Department. Vendor maintenance work on reactor trip system components must comply with applicable plant procedures. Vendor implementing procedures prepared in accordance with the vendors QA program which must be approved by FPL QA may also be used.

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

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 informationhandling 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.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

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 action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgment for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Response

1.1 Quality Instruction QI 2-PR/PSL-1 (Quality Assurance Program) outlines the scope of those items and activities used to develop and implement the St. Lucie Unit 1 and 2 Quality Assurance Program to assure conformance to the requirements in 10 CFR 50, Appendix B and the FPL QA manual. This QI specifies that the extent of quality requirements may be determined by review of documents such as the FSAR, Piping Line and Valve Lists, Instrument Lists, Drawings, original equipment specifications, NFPA code, CFRs and associated Codes or Specifications. Appendix A to this Quality Instruction provides a scope of those activities which have been determined as requiring one or more aspects of the PSL Quality Program. This appendix is attached.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

Section 3.2 of the Updated Final Safety Analysis Report for Unit 1 and the FSAR for Unit 2 gives examples of interfaces between the non-seismic and the seismic portions of systems. It also states that the interfaces between non-seismic and seismic systems are depicted on the P&I Diagrams. These drawings are used primarily to classify components as safety-related.

1.2 Safety-related components are identified in the FSAR, P&I Diagrams, Valve Lists, and Instrument Lists. These lists and drawings were developed during plant design and construction and are now maintained as controlled documents. The approval authority for revising specific technical, FSAR and PC/M items impacting the quality program is vested in the FPL engineering organization.

1.3 Reviews of work documents and plant change/modifications packages are conducted by quality trained personnel for assurance of proper classification. The above-mentioned documents are used in the reviews for component classification. In addition, the FPL Quality Procedures as contained in the FPL Quality Assurance Manual are followed to ensure that procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B apply to safety-related components. The most recent revision to the FPL Topical Quality Assurance Report was found to be acceptable by the NRC on September 7, 1983.

1.4 Like all aspects of the QA Program, procedures for preparation, validation and routine utilization of the safety-related identification documents are the subject of routine Quality Assurance audits.

1.5 FPL Quality Procedure 4.1 provides a system to assure that the appropriate technical and quality requirements are placed upon suppliers who provide material, equipment and services for FPL nuclear plants. A copy of this Quality Procedure is attached.

1.6 No response required.

2.0 FPL is a member of the Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2 formed on September 1, 1983, for the specific purpose of defining an appropriate vendor interface program. At present we intend to incorporate the result of the NUTAC. Our schedule for submission of our program description was provided in our letter L-83-480, dated September 7, 1983. This report is scheduled for submittal by February 29, 1984.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

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 Specifications, where required.

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

Response

1. FPL has reviewed test and maintenance procedures and technical specifications and has determined that current maintenance procedures provide for retest/functional operation of equipment after maintenance is performed. Technical Specification 6.8.1 requires that these procedures be implemented.

2. All identified recommendations concerning the reactor trip switchgear have been included in appropriate test and maintenance procedures.

   As discussed in Section 2.1, the Availability Data Program and the CE Infobulletins provide information and recommendations on CE NSSS equipment. Future changes to test guidance would show up in this program and would be dispositioned by means of the Operating Experience Feedback Program.

3. FPL has not identified any post maintenance test requirement in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Should such requirements be discovered in the future, changes to the requirements will be submitted for staff approval.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

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-maintenace test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Response

1. Test and maintenance procedures have been reviewed and we have determined 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. Technical Specification 6.8.1 requires that these procedures be implemented.

2. All identified engineering and vendor recommendations have been included in the test and maintenance procedures. Again, for Combustion Engineering supplied equipment, these recommenations are provided in Infobulletins as part of the CE Availability Data Program.

   In addition, as participants in the INPO See-In Program, significant events occurring throughout the nuclear industry and important vendor information items are entered in the FPL Operating Experience Feedback Program. This program provides additional assurance that test and maintenance items which have caused problems at other plants will be reviewed for applicability to St. Lucie Units 1 and 2 and dispositioned.

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

3. FPL has not identified any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Should such test requirements be discovered in the future, changes to the requirements will be submitted for staff approval.

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

All identified reactor trip circuit breaker modifications have been implemented.

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.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

Response

1. Maintenance procedure 1-0110060 (Periodic Maintenance of Control Element Assembly /CEA) Drive Equipment and Switchgear) provides guidelines for the periodic maintenance of the reactor trip switchgear. This procedure has been reviewed and has been determined to meet or exceed the vendors recommended guidelines.

2. The current procedure mentioned above checks the torque required on the trip shaft to trip a closed breaker. If the torque falls between 20 and 24 ounce-inches, revitalization of the lubrication is in order. By maintaining a margin between the 20 ounce-inch torque valve and the 24 ounch-inch maximum value, degradation of operability is prevented. A revision has been prepared to this test procedure to record the time required for the breaker to trip. This change will provide a readily available means to forecast degradation of operability. This procedure is now undergoing plant review prior to approval and implementation.

3. An alternative to a life testing program for the breakers (including the trip attachments) is being considered by the Combustion Engineering Owners Group (CEOG) of which FPL is a member. The CEOG Steering Committee has requested that the Analysis Subcommittee review the B&W approach for addressing trip breaker life cycle testing and the BWOG request that the CEOG join in this program. A status of this project should be available following the December Steering Committee meeting.

4. The periodic replacement of breakers or components will be considered upon completion of the owners group activities discussed above.

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

This item is not applicable to St. Lucie Units 1 and 2.

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

This item is not applicable to St. Lucie Units 1 & 2.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

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 Action 4.3 above) and CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants (see Action 4.4 above); and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

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

3. Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when acccounting 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. On-line functional testing of the diverse trip features, including the breaker undervoltage and shunt trip features, are performed at least monthly in accordance with Operating Procedure 1400059 (Title - Reactor Protection System - Periodic Logic Matrix Test). The channel functional test of the Reactor Protection System Logic and Reactor Trip Breakers is required monthly by Technical Specifications 4.3.1.1.1 (Unit 1) and T.S. 4.3.1.1 (Unit 2) and Table 4.3-1.

ATTACHMENT

Re: St. Lucie Units 1 & 2
Docket Nos. 50-335, 50-389
Generic Letter 83-28

2. Not applicable.

3. As a member of the Westinghouse Owners Group, FPL is participating in the Technical Specification Optimization Program. This program has documented an evaluation of the impact on RPS unavailability of current and extended surveillance intervals. The WCAP generated by this program is undergoing staff review. If the WOG program is acceptable to the NRC and if extended surveillance intervals are determined to be acceptable to the NRC, then FPL will attempt to initiate a similar review by the CE Owners Group, with the goal of determining the optimal surveillance intervals for St. Lucie Units 1 & 2.

   The CEOG Steering Committee has requested a proposal from C-E for addressing the NRC Salem ATWS requirement to justify reactor trip system surveillance testing intervals. A schedule is not available at this time.