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50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251

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SUBJECT: Responds to NRC 790917 ltr re integration between safety-grade sys & nonsafety-grade sys. Forwards descriptions of four potential interaction scenarios & generic vendor recommendations justifying continued operation.

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FLORIDA POWER & LIGHT COMPANY

October 8, 1979
L-79-284

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

Dear Mr. Denton:

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Safety/Control Interactions

Florida Power & Light Company (FPL) has reviewed your letter of September 17, 1979 on the subject of interaction between safety grade systems and non-safety grade systems. The following information is submitted in response to your letter, and provides a basis for continued operation of Turkey Point Units 3 and 4. The basis is founded primarily on the improbability of the postulated scenarios as they apply to the Turkey Point units, the acceptability of the consequences, and both short-term and long-term commitments to resolve the issue.

On September 18, 1979 Westinghouse presented to the NRC Staff a summary of the investigation that led to the identification of four potential interaction scenarios where the affect on control systems of adverse environments (resulting from high energy line breaks) could lead to consequences more limiting than the results presented in the Safety Analysis Report. Table 1 of Appendix B summarizes the scope of the Westinghouse investigation. The accidents considered encompass all postulated High Energy Line Break (HELB) environments, including all break locations and a range of break sizes. Of 49 combinations of control system and accident environment investigated, 15 interaction scenarios (denoted by an X in Table 1) were identified which resulted in consequences less conservative than reported in the Safety Analysis Report. However, the 15 interactions are bounded by the four interactions discussed in the attachments. These four interactions are evaluated in Appendix A, which is an evaluation performed by FPL specific to Turkey Point Units 3 and 4. The conclusion of the evaluation is that no significant safety hazard is posed and continued safe operation of the units is justified. Additional efforts directed toward long-term solutions are underway. Furthermore, Appendix B (with Attachments I - IV) is an evaluation performed by the NSSS vendor which considers the four interactions with respect to probabilities for occurrence and potential consequences. This evaluation also concludes that continued safe operation of the units is justified until final resolutions are implemented. Attachments I - IV describe the four interactions and include generic recommendations made by the NSSS vendor. These recommendations either have been reviewed or are being reviewed by FPL for applicability, and for the determination of alternatives for final resolutions.

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FPL is attempting to resolve the four items identified in this submittal within the following schedule:

- items whose solutions do not require equipment modifications (i.e., procedural changes, training, etc.) are scheduled for resolution by January 1, 1980.

- items whose solutions require equipment modifications are scheduled for resolution by June 1, 1981.

Additionally, FPL will continue an on-going program of investigation into other potentially similar interaction mechanisms.

Very truly yours,

E. A. Adomat

for Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/MAS/RJA/cph

Appendices (2)

cc: Mr. James P. O'Reilly, Region II
Robert Lowenstein, Esquire

APPENDIX A

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Safety/Control Interactions

POTENTIAL IMPACT ON BOP SYSTEMS
DUE TO ADVERSE ENVIRONMENT RESULTING
FROM HIGH ENERGY LINE BREAKS

Westinghouse has considered the interaction of Balance of Plant (BOP) systems as part of their continuing review of the environmental qualifications of Westinghouse supplied NSSS equipment. Their review is being performed to determine if performance of BOP equipment (not presently environmentally qualified) may impact the protective functions performed by NSSS safety grade equipment.

As a result of the review, the following four systems have been identified as potentially susceptible to undesirable control system operation induced by an adverse environment (See Appendix B NSSS information). These systems could potentially malfunction if impacted by the adverse environment resulting from a high energy line break inside or outside containment.

1) Steam Generator Power Operated Relief Valve (PORV) Control System

2) Main Feedwater Control System

3) Pressurizer PORV Control System

4) Rod Control System

The Westinghouse evaluation identified these concerns as not constituting a substantial safety hazard. Further plant specific reviews to determine the applicability (or inapplicability) to individual units was recommended and were conducted by FPL. The status of these reviews is as follows:

1) S/G PORV Control System

This concern is not a problem at Turkey Point because the control systems are located in open areas (i.e., outdoors) such that direct impingement resulting from a feedline rupture, or a significant change in the temperature of the area, are not feasible.

2) Main Feedwater Control System

This concern requires additional consideration because the feedflow transmitters are located relatively close togehter in the vicinity of a feedline. There is, however, a very low probability of occurrence, because the transmitters are located in an open area (i.e., outdoors) such that direct impingement would be required in order to produce an adverse environment that could possibly result in undesirable effects. This would require a break size and location precisely oriented to impinge on the three transmitters. Such a specific break is a highly unlikely event. Moreover, should this highly unlikely event occur, operator action would be taken prior to all three steam generators reaching the low-low level trip setpoint (see Appendix B scenarios), thereby mitigating the consequences of the event. Additional evaluation of this event is planned.

3) Pressurizer PORV Control System

A feedline break inside containment may affect the environment in the building, thereby subjecting the PORV control systems to elevated temperatures and possibly causing them to open. However, the following circumstances provide reasonable assurance of mitigating any possible effects resulting from the unlikely occurrence of this event:

(a) block valves are available to isolate an open pressurizer PORV

(b) operators are aware of the symptoms and required procedural actions for stuck open PORV's (result of TMI reviews and training)

Additional evaluation of this event is planned.

4) Rod Control System

A steamline break inside containment may subject the excore detectors and cables to elevated temperatures which could cause rod withdrawal if the rods are in the automatic control mode prior to reactor trip. However, the nuclear instrumentation input signal to the rod control system has been tagged out. This effectively eliminates the possibility for occurrence of this event until a final resolution is implemented. Additional evaluation of this event is planned.

APPENDIX B

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Safety/Control Interaction

INTERACTION SCENARIOS

Probability of Postulated Interactions

Implicit in the four (4) potential interaction scenarios identified by Westinghouse are worst case assumptions concerning the break size and location, and the type and extent of consequential failures in control systems induced by the adverse environment. These assumptions are therefore in addition to the already conservative set of assumptions ascribed to the analysis of the Design Basis Events reported in the Safety Analysis Report. It follows that these scenarios represent a significantly less probable subset of the Design Basis Events that are dependent on the occurrence of additional events, each having an associated uncertainty of occurring. While no quantitative analysis has been conducted concerning the improbability of overall scenarios, the attachments define, for each of the scenarios considered as applicable to Turkey Point, the conservative assumptions already contained in the Design Basis Event analysis reported in the Safety Analysis Report and the additional conservative assumptions to be made to derive the postulated interaction scenario.

As can be seen from the attachments, for each of the scenarios considered, the improbability of all the additional set of assumed conditions occurring simultaneously, over and above the already low probability of the Design Basis Event itself, leads to the conclusion that continued operation of Turkey Point can be justified until the proposed schedule for implementing solutions to these low probability event scenarios can be implemented.

With regard to the probability of any single design basis event initiating, via the adverse environment, failures in several control systems, it again can be noted from the attachments that the probability of all the additional set of conditions occurring simultaneously for more than one scenario is of an even lower order of magnitude than for each individual scenario. Furthermore, implementation of the proposed long term solutions for the individual scenarios will, as a consequence, address any concern for multiple interactions from a single initiating Design Basis Event.

Due to the implementation in the design of the electrical separation requirements between control and protection systems specified in IEEE-279, the only interaction mechanisms identified in the above scenarios result from conservatively assuming an adverse environment at the location of the control systems and the consequential equipment failure in the worst direction. As a consequence, it can be anticipated that any interaction scenarios yet to be identified, in as yet unreviewed control systems, will be no more probable than the particular scenarios described by Westinghouse.

Consequences of Postulated Interactions

In lieu of performing a plant specific analysis in an effort to address each of the potential postulated interactions involving a feedline break, Westinghouse has referred to bounding accident analyses that have been submitted to the NRC in WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS. Section 4.2 of the report provides transient results following a total loss of main and auxiliary feedwater. Sensitivity studies as a function of time of auxiliary feedwater initiation and opening of the pressurizer power operated relief valves are presented following the initial transient. Calculations have been performed to show that the consequences following the control interaction scenarios for the steam generator PORV control system, main feedwater control system and pressurizer PORV control system are in fact bounded by the analyses in WCAP-9600. For all accident scenarios, the calculations indicated that the operator need not take corrective action to mitigate the consequences for at least 30 minutes following initiation of the event.

A typical analysis has been performed to address the rod control system interaction scenario. The results of the analysis indicate that no fuel damage occurs and the consequences are within the assumptions made in the Safety Analysis Reports.

Recommended Solutions

Both short and long term solutions are recommended to the utilities in the attachments. The short term recommendations for the steam generator

PORV, main feedwater and pressurizer PORV control systems involve revisions to the Westinghouse Emergency Operating Instructions. The short term solution for the rod control interaction scenario is to refer to the results of the typical analysis performed by Westinghouse showing no fuel damage.

Control System / Accident	Reactor Control	Pressurizer		Feedwater Control	Steam Generator Pressure Control	Steam Dump System	Turbine Control
		Pressure Control	Level Control				
Small Steamline Rupture	X	X			X		
Large Steamline Rupture		X			X		
Small Feedline Rupture	X	X		X	X		
Large Feedline Rupture	X	X			X		
Small LOCA	X	X		X			
Large LOCA							
Rod Ejection							

TABLE 1

PROTECTION SYSTEM-CONTROL SYSTEM POTENTIAL ENVIRONMENTAL INTERACTION

X - POTENTIAL INTERACTION IDENTIFIED THAT COULD DEGRADE ACCIDENT ANALYSIS

☐ - NO SUCH INTERACTION MECHANISM IDENTIFIED

ATTACHMENT I

STEAM GENERATOR PORV CONTROL SYSTEM

1. Summary of Postulated Scenario

Following a feedline rupture outside containment in the auxiliary building, the steam generator PORV's are assumed to exhibit a consequential failure due to an adverse environment. Failure of the PORV's in the open position results in the depressurization of multiple steam generators which are the source of steam supply for the turbine driven auxiliary feedwater pump. Eventually, the turbine driven auxiliary feedwater pump will not be capable of delivering auxiliary feedwater to the intact steam generators. Depending upon auxiliary system design, a potential exists that no auxiliary feedwater will be injected into the intact steam generators until the operator takes corrective action to isolate the auxiliary flow spilling out the rupture.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break

- conservative initial assumptions
  - Appendix K decay heat model
  - Engineered safeguards power plus calorimetric error
  - programmed RCS temperature plus control deadband and instrument errors
  - initial conservative S/G inventory
  - conservative core physics
- conservative accident assumptions
  - break (all sizes) in Safety Class 2 feedline piping
  - maximum adverse environmental errors for protective instrumentation
  - worst single active failure
  - operator action time.

b. Additional Assumptions Required for this Scenario

- break must occur outside containment between the penetration and feedline check valve.

- Adverse environment resulting from the rupture can impact the steam generator PORV control systems associated with the ruptured loop and the intact loops.

- The single active failure is a motor driven auxiliary feed pump. The loss of a turbine driven auxiliary feed pump as the single active failure or no active failure would invalidate the postulated scenario.

- Due to the adverse environment, the steam generator PORV control system initiates a spurious signal to open the PORV(s). Should the control system continue to operate within specification or initiate a spurious signal to close the PORV(s) the scenario is invalidated.

- PORV on steam generators supplying steam to turbine driven auxiliary feed pump is assumed to open as a result of spurious signal. If this PORV is not affected or fails closed, the scenario is invalidated.

3. Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems, describes transient analyses for postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that the operator has at least 4,000 seconds following the loss of all feedwater to reinitiate auxiliary feedwater flow to the steam generators before the core begins uncovering.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The only additional assumption made is that a feedline rupture occurs outside containment between the containment penetration and the feedline check valve. Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), calculations have shown that the heat removal capability of the liquid inventory blowdown requires operator action 1200 seconds earlier than reported in WCAP-9600. Thus, if a feedline rupture is assumed coincident with the analyses performed in WCAP-9600 the operator still has at least 2800 seconds to take corrective action to inject auxiliary feedwater into the intact steam generators. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

4. Recommended Short Term Solution

The operator should be alerted to the possibility of the steam generator PORV's failing in the open position following a secondary high energy line rupture outside containment in the auxiliary building. If control of the block valves in the steam generator PORV relief lines is possible from the control room, a caution should be added to the secondary high energy line break emergency operating instruction directing the operator to close the block valves. If the block valves can only be operated locally, the operator should be cautioned that the steam driven turbine auxiliary feedwater pump could potentially be lost due to loss of steam supply. If that occurs, the operator can only rely upon the motor driven auxiliary feedwater pumps to supply the minimum auxiliary feedwater requirements following a secondary line rupture.

Other than the caution to the operator discussed above, the actions that must be taken by the operator that are currently recommended in the Westinghouse Reference Operating Instructions, continue to be applicable. No additional actions are required to mitigate the consequences of the accident.

5. Recommended Long Term Solution

The long term hardware solution involves the addition of two qualified solenoid valves per steam generator PORV. These redundant (Train A and B) solenoids will ensure that the PORV is vented following a steam or feedline break to prevent spurious opening of a PORV due to a control system malfunction. The protection grade block logic for the solenoids is initiated from a steam line isolation signal. Means are also provided to the operator to unblock the air supply for use when proceeding to cold shutdown. This relatively simple solution is possible since it is only necessary to prevent the opening of the PORV.

ATTACHMENT II

MAIN FEEDWATER CONTROL SYSTEM

1. Summary of Postulated Scenario

Following a small feedline rupture the main feedwater control system malfunctions in such a manner that the liquid mass in the intact steam generators is less than for the worst case presented in Safety Analysis Reports. The reduced secondary liquid mass at time of automatic reactor trip results in a more severe reactor coolant system heatup following reactor trip.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break

- conservative initial assumptions
  - Appendix K decay heat model
  - Engineered safeguards power plus calorimetric error
  - Programmed RCS temperature plus control deadband and instrument error
  - initial conservative S/G inventory
  - conservative core physics

- conservative accident assumptions
  - break (all sizes) in Safety Class 2 feedline piping
  - maximum adverse environmental errors for protective instrumentation
  - worst single active failure (loss of any one auxiliary feed pump)
  - operator action time

b. Additional Assumptions Required for this Scenario

- break must occur between S/G nozzle and feedline check valve. A break at any other location invalidates the scenario.

- Small breaks less than 0.2 sq ft. Larger breaks invalidate the scenario.

- Adverse environment resulting from the break can impact both the main feedwater control systems associated with the broken loop and the intact loops.

- Due to the adverse environment the main feedwater control system initiates a spurious signal to close the feedwater control valves (FCV) in the intact loops. Should the control system continue to operate within specification the scenario is invalidated.

3. Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS System, describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). Following a loss of all main and auxiliary feedwater, the operator is not required to take action for at least 4,000 seconds following the loss of all feedwater to prevent the core from uncovering. With a feedline rupture assumed coincident with the assumptions made in WCAP-9600, the operator continues to have at least 2800 seconds before corrective action must be taken to inject auxiliary feedwater into the intact steam generators to prevent core uncovering. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

4. Recommended Short Term Solution

To ensure that the operator is aware of this possible control system environmental interaction, the system transient characteristics following a small feedline rupture with and without feedwater control system operation should be reviewed by the operator.

The general system characteristics following a small feedline rupture would be the following: a slowly decreasing indicated water level in at least one steam generator, a resultant opening of the associated feedwater control valve, and a corresponding increase in main feedwater flow. One or more of the above trends would be indicative to the operator that a small feedline rupture has occurred.

If, in addition, a main feedwater control valve was assumed to close in a loop with a decreasing steam generator water level due to a control system environmental interaction, the abnormal operating characteristic of the feedwater control system would be immediately apparent to the operator. After observing the abnormal operating characteristics, the operator would immediately initiate corrective action to restore main feedwater flow and if not successful, manually trip the reactor. Provided that the operator manually trips the reactor berfore the secondary liquid inventory is less than that assumed in the analysis, the Safety Analysis Report licensing basis is met.

5. Recommended Long Term Solution

The solution to this problem may be accomplished by either of the following: revising the criteria or adding additional auxiliary feedwater pumping capacity.

Revision of the criteria could be used by accepting hot leg saturation in the reactor coolant system prior to transient turnaround. Alternatively adding additional auxiliary feedwater flow capacity would overcome the need for assuming operation of the main feedwater control system.

ATTACHMENT III

PRESSURIZER PORV CONTROL SYSTEM

1. Summary of Postulated Scenario

Following a feedline rupture inside containment, the pressurizer PORV control system malfunctions in such a manner that the power operated relief valves fail in the open position. Thus in addition to a feedline rupture between the steam generator nozzle and the containment penetration, a breach of the reactor coolant system boundary has occurred in the pressurizer vapor space.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break

- conservative initial assumptions
  - ° Appendix K decay heat model
  - ° Engineered safeguards power plus calorimetric error
  - ° Programmed RCS temperature plus control deadband and instrument errors
  - ° initial conservative S/G inventory
  - ° conservative core physics

- conservative accident assumptions
  - break (all sizes) in Safety Class 2 feedline piping
  - maximum adverse environmental errors for protective instrumentation
  - worst single active failure (loss of any one auxiliary feed pump)
  - operator action time

b. Additional Assumptions Required for this Scenario

- break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario.

- double ended break leads to limiting consequences. Smaller breaks permit longer operator action times.

- adverse environment resulting from the break can impact the pressurizer power operated relief valve control system.

- due to the adverse environment the pressurizer PORV control system initiates a spurious signal to open the PORV(s). Should the control system continue to operate within specification or initiate a spurious signal to close the PORV's the scenario is invalidated.

- should the PORV's fail to the preset safe position (i.e. closed) the scenario is invalidated.

3. Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems, describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that, in the event that the operator cannot restore auxiliary feedwater to the steam generators, the operator is required to open the pressurizer PORV's within 2,500 seconds to maintain adequate core coolant inventory.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The additional assumptions made are the following:

a. a feedline rupture is assumed to occur between the steam generator nozzle and the containment penetration

b. auxiliary feedwater is injected into the intact steam generator following the feedline rupture.

Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), the loss of heat sink due to the liquid inventory blowdown of the ruptured steam generator is more than counterbalanced by the auxiliary feedwater being injected into the intact steam generators following reactor trip. Therefore, the results of the analyses present in WCAP-9600, Section 4.2, which illustrates that the operator is not required to take corrective action for at least 2,500 seconds following the loss of feedwater also applies to this scenario. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

4. Recommended Short Term Solution

The operator should be alerted to the possibility of the pressurizer PORV's failing in the open position following a high energy line rupture inside containment. After identifying a high energy line rupture inside containment, the operator should be instructed to close the block valves in relief lines of the pressurizer PORV's. Closure of the block valves will ensure that a secondary high energy line rupture inside containment will not result in a breach of the primary pressure boundary integrity. The Westinghouse Reference

Operating Instructions already instruct the operator to close the pressurizer PORV's after a primary high energy line rupture is diagnosed.

After the operator closes the PORV relief line block valves, the actions recommended in the Westinghouse Reference Operating Instructions continue to be applicable. No additional actions are required to mitigate the consequences of this scenario.

5. Recommended Long Term Solution

An acceptable solution to this problem is to demonstrate that the control system, including the PORV's, will operate normally or fail the valve closed following a high energy line rupture inside the containment. An alternate solution is to provide an additional motor operated valve (MOV) in each line and capability to close both from protection grade logic initiated by a protection grade signal.

Two additional solutions involve upgrading the PORV to an active valve and demonstrating that it will close when required under adverse conditions. The first involves using the same logic to close the existing MOV and vent the PORV and the second requires adding two safety grade solenoids (Train A and B) to vent the PORV. The latter solution is preferred if the PORV can be upgraded, if not, then two qualified MOV's should be provided in each line.

ATTACHMENT IV

ROD CONTROL SYSTEM

1. Summary of Postulated Scenario

Following an intermediate steamline rupture inside containment, the automatic rod control system exhibits a consequential failure due to an adverse environment which causes the control rods to begin stepping out prior to receipt of a reactor trip signal on overpower delta-T. This scenario results in a lower DNB ratio than presently presented in Safety Analysis reports.

2. Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Steamline Break

- conservative initial assumptions
  - nominal rated power plus calorimetric error
  - Programmed RCS temperature plus control deadband and instrument errors
  - conservative end of life core physics

- conservative accident assumptions
  - break (all sizes) in Safety Class 2 steamline piping
  - maximum adverse environmental errors for protective instrumentation
  - worst single active failure (loss of any one Safety Injection pump)
  - operator action time

b. Additional Assumptions Required for this Scenario

- break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario.

- intermediate steamline breaks (0.1 to 0.25 sq. ft. per loop) at power levels from 70 to 100 percent. Other break sizes and power levels invalidate the scenario.

- adverse environment from the break can impact the nuclear instrumentation system (NIS) equipment (i.e. excore neutron detectors, cabling connectors, etc.) prior to reactor trip (i.e. within 2 minutes).

Should the NIS equipment not be affected until after reactor trip (i.e. later than 2 minutes) the scenario is invalidated.

- due to the adverse environment the NIS system initiates a spurious low power signal without causing a reactor trip on negative flux rate. Should the NIS continue to operate within specification, initiate a spurious high power signal or cause a reactor trip on negative power rate the scenario is invalidated.

3. Accident Consequences

A typical bounding analysis of the intermediate steamline rupture was performed to calculate the extent of fuel damage due to rod control system withdrawal prior to reactor trip. Based upon the reduction in radial peaking factor with burn-up and conservative end-of-life physics parameters, no fuel damage was calculated to occur following the intermediate steamline rupture with a consequential rod control system failure.

4. Recommended Short Term Solutions

- As discussed above, a generic intermediate steamline rupture inside containment which results in control rod withdrawal due to a control system environmental interaction prior to reactor trip was analyzed. The results of the analysis indicated that no fuel damage occurred, which is consistent the assumptions made in the applicable Safety Analysis Reports.

An alternate short term solution would be to commit to "at power" operation with the rod control system in manual control, relying upon the operator to maintain desired steady state conditions.

5. Recommended Long Term Solutions

A plant specific analysis could be performed which may show no fuel damage occurs as a result of this scenario. Alternately, a hardware solution to this problem requires the qualification of the excore detectors and associated in-containment equipment to the bounding steamline break envelope. This ensures proper operation of the control system following such an incident. If this proves impractical, these detectors should be qualified to a steamline break envelope of 200°F and analysis done to show that a high containment pressure trip point is reached prior to exceeding 200°F inside containment. The containment pressure trip function then becomes the primary trip function with the overpower delta T function the diverse back-up.

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

E. A. Adomat, being first duly sworn, deposes and says:

That he is Executive Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee

E. A. Adomat

E. A. Adomat

Subscribed and sworn to before me this

8 day of October, 1979

Theresa M. Miranda

NOTARY PUBLIC, in and for the County of Dade,
State of Florida

My commission expires: May 5, 1981

NOTARY PUBLIC STATE OF FLORIDA at LARGE
COMMISSION EXPIRES MAY 5 1981
THRU MAYNARD BONDING AGENCY