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SUBJECT: Forwards response to 880223 request for addl info re Generic Ltr 86-06, "Reactor Coolant Pump Trip Criteria." CEN-268 revised to delete any ref to use of containment radiation alarms as possible criteria for tripping second two pumps.

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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Generic Letter 86-06
Reactor Coolant Pump Trip Criteria

By letter dated February 23, 1988, the NRC requested additional information related to Florida Power & Light Company's (FPL) reactor coolant pump trip criteria at St. Lucie Units 1 and 2. FPL had responded to Generic Letter 86-06 by letter L-87-265 dated July 1, 1987.

Attached is FPL's response to this request for additional information. If additional discussion is required, please contact us.

Very truly yours,

W. F. Conway

W. F. Conway
Acting Group Vice President
Nuclear Energy

WFC/EJW/gp

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

EJW86-06.RCP

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ATTACHMENT 1
FPL RESPONSE TO ADDITIONAL QUESTIONS
RE GENERIC LETTER 86-06
IMPLEMENTATION REQUIREMENTS

Question Number 1

Florida Power & Light's response to Generic Letter 86-06, Item 1, presented a strategy for tripping the second two pumps that was different from that recommended in CEN-268. CEN-268 recommended the second set of pumps be tripped if one of the following criteria was met:

a. Subcooling less than 20°F and a containment radiation alarm.

b. Subcooling less than 20°F and no secondary radiation alarm.

c. Subcooling less than 20°F, a containment radiation alarm, and no secondary radiation alarm.

The strategy presented for St. Lucie, Units 1 and 2, called for tripping the second set of pumps if the secondary pressure remained above 750 psia and the containment pressure was above 4 psig or the secondary radiation alarms did not actuate.

Clarify how these parameters and setpoints are used in the plant specific pump trip guidelines. Because the strategy proposed for tripping the second set of pumps at St. Lucie, Units 1 and 2, is different from that recommended in CEN-268, Florida Power & Light must show that this strategy is as capable of discriminating between LOCA and non-LOCA events as that presented in CEN-268. That is, it must be shown that the St. Lucie, Units 1 and 2, strategy will trip the second two pumps for all small breaks where pump trip is necessary to prevent fuel clad temperatures from exceeding licensing limits but will not trip the second set of pumps for steam generator tube ruptures, steam line breaks, and increased heat removal anticipated operational occurrences. In addition, for small breaks LOCAs, it must be shown the St. Lucie, Units 1 and 2, strategy will trip the second set of pumps in the same time period as the criteria presented in CEN-268.

Response

CEOG-87-269, "Revision to CEOG RCP Trip Two/Leave Two Report (CEN-268)", revises CEN-268, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients", to delete any reference to the use of containment radiation alarms as possible criteria for tripping the second two RCPs.

FPL's strategy on Trip Two/Leave Two is functionally equivalent to the recommendations of CEN-268. The method employed by FPL for LOCA determination is mirrored from Figure 5-2, "Break Identification Chart", from the NRC approved CEN-152, Revision 3, "Combustion Engineering Emergency Procedure Guidelines." [FPL utilizes CE's Break Identification Chart in the St. Lucie Plant Emergency Operating Procedures (EOPs) nearly verbatim in the 'Break Identification Chart' and in principle with the 'Diagnostic Flow Chart']. The following are calculations for justifying this equivalency:

CALCULATIONS AND CONCLUSIONS REGARDING THE PARAMETERS USED TO TRIP RCP'S

A. RCS PRESSURE: 1300 PSIA

The best estimate of the pressure difference between the LOCA RCS pressure plateau and the secondary pressure is about 50 psia. Also, as shown in CEN-268, Supplement 1-P, secondary pressures stabilize (at either the SBCS setpoint of 900 psia or the lowest MSSV lift point of 1000 psia) at about the same time as the RCS pressure plateau is reached. It is therefore reasonable to state that the realistic values of the RCS pressure plateau are 950 and 1050 psia, respectively. It has also been shown that a conservative estimate of the pressurizer pressure error (for both St. Lucie Units 1 and 2) is about +192 psia. Adding this error to the plateau values gives 1142 and 1242 psia, respectively. Both values represent a substantial margin relative to the EOP value of 1300 psia, so that tripping the first 2 RCP's at an indicated value of 1300 psia accomplishes the desired goal of protecting the remaining RCS inventory for small break LOCA's.

B. STEAM GENERATOR PRESSURE: 800 psia (old value) 750 psia (new value)

This value in the Diagnostic Flow Chart represents a 100 psia drop in S/G pressure below the lowest expected S/G pressure (900 psia via SBCS) if an Excess Steam Demand (ESD) is not occurring. It has been calculated that the S/G pressure error under adverse conditions (about 113 psia) slightly exceeds the margin of 100 psia. Recognizing this, and recognizing that secondary pressure is also influenced by Main Feedwater and (possibly) Auxiliary Feedwater following a reactor trip, a revised value of 750 psia is now used. This selection bounds the error calculation, is still well above the MSIS setpoint of 600 psia, and has been rounded off for operator use. In summary, pressurizer level and pressure decreasing concurrent with a secondary pressure of less than 750 psia is used to diagnose an ESD. Secondary pressure greater than 750 psia concurrent with containment pressure less than 4 psig and pressurizer level and pressure decreasing is the criteria used to diagnose a Steam Generator Tube Rupture (SGTR). Presence of secondary steam radiation is used to confirm this diagnosis. Absence of secondary steam radiation will indicate a possible LOCA outside of containment (i.e., letdown line break).

C. CONTAINMENT PRESSURE: 2 psig (old value) 4 psig (new value)

The value in the Diagnostic Flow Chart represents a pressure which plant operators recognize as being above the Technical Specification value of 1.8 psig. However, the error of 1.6 psi when added to even 0 psig gives a result too close to 2 psig to be within normal resolution capabilities. Adding the error to the Technical Specification value gives 3.4 psig, which exceeds 2 psig. Accordingly, 2 psig has been changed to 4 psig in the Diagnostic Flow Chart.

It should be noted that this change, does not significantly affect the containment temperatures (135^{o}F, 175^{o}F) corresponding to a 2 psi increase in containment pressure. This is because the calculated values assumed an initial temperature of 120^{o}F, which is the Technical Specification limit. Since the calculations are incremental in nature, it is the increases in containment pressure which result in the increases to an initially assumed high containment temperature. In other words, 4 psig can be a pressure increase from 4.0 - 1.8 = 2.2 psig up to an increase of 4.0 - 0.0 = 4.0

psig. This means that a potential increase of 4 psig could cause temperatures above 135°F and 175°F, respectively. For a LOCA, the 4 psig value would still be less than 175°F as shown earlier. For an ESD at a 4 psig increase, the temperature would exceed 175°F. This process, if continued past this point, would increase the calculated adverse errors beyond those already determined for pressurizer and steam generator pressures; it would have no effect on the containment pressure errors since these transmitters are located outside of containment for both units. Recognizing that: 1) the LOCA 4 psig temperature value is still less than 175°F, 2) the LOCA 1300 psia related temperature is still less than 175°F, 3) it is not likely that an ESD will be misdiagnosed as a LOCA and that plant safety is not challenged even if all RCP's are tripped for this situation, and 4) the thermal lag of the instrumentation has not been credited in any of the adverse error calculations, it is concluded that there is no need to revise the calculated adverse errors based on using 4 psig versus the current 2 psig. Finally, in this context it should also be noted that in a strictly mechanical sense per the Diagnostic Flow Chart, 2 (or 4) psig may be reached before the RCS pressure plateau is reached for LOCA's in containment. Hence, the margin inherent in tripping the first 2 RCP's at 1300 psia is reinforced very early by the diagnostics to enter the LOCA procedure and trip all operating RCP's. These considerations add further weight to the already solid argument that the existing adverse errors have been conservatively calculated at 175°F and that further iterations are not needed.

In summary, pressurizer pressure and level decreasing concurrent with containment pressure greater than 4 psig and secondary steam pressure greater than 750 psia are the criteria used to diagnose a LOCA inside of containment.

D. <u>Conclusions</u>

The T2/L2 strategy meets the NRC analytical guidance for justification of manual RCP trip. An analysis of the worst small break LOCA with the licensing analysis demonstrates compliance with 10CFR50.46 and Appendix K limits. A most probable best estimate small break LOCA analysis was conducted to show that there is <u>no</u> required time limit for operator action to terminate RCP operation. This analysis satisfies ANSI Standard 58.8 for the minimum time for operator action.

CEN-268 provides guidance regarding a decision process for tripping the second set of reactor coolant pumps based on RCS subcooling and steam plant radiation. The overriding philosophy of CEN-268, however, is that the second two (2) pumps should be tripped when a LOCA is diagnosed by whatever means available. Therefore, FPL's choice to employ CEN-152's criteria for LOCA determination is acceptable as a decision point for tripping the second set of pumps.

<u>Question Number 2</u>

Florida Power & Light did not provide sufficient information in its July 1, 1987 response to GL 86-06 Item 3 to determine how the uncertainties in the generic analysis presented in CEN-268 affect the results as they apply to St. Lucie, Units 1 and 2. This is especially true because Florida Power & Light chose to ignore the CE recommended setpoint of 1210 psia for tripping the first two pumps. In its discussion on why the CE recommended setpoint was not chosen, Florida Power & Light referred to less conservative analyses that gave pressure plateaus of 950 and 1050 psia. These less conservative analyses were not discussed in CEN-268 or CEN-268, Supplement 1. In order to justify the 1300 psia setpoint, the licensee must compare the values of the parameters used by CE in the conservative analysis establishing the 1210 psia setpoint to the values of these same parameters in the less conservative analyses referred to by Florida Power & Light. The plant specific values for these parameters should also be provided. At a minimum, Florida Power & Light should discuss core power; decay heat; HPIS capacity; makeup flows; setpoints for steam generator safety valves; and setpoints for reactor trip, safety injection, and accumulator injection. Show that the values used in the less conservative analyses are either representative of those at St. Lucie, Units 1 and 2, or conservative. If a reference plant parameter is not representative for St. Lucie, Units 1 and 2, discuss how this was considered in determining the plant specific setpoints.

Response

FPL did not ignore CEN-268's generic setpoint of 1210 psia; but rather chose to employ more realistic plant specific parameters in determining RCS pressure plateaus. These values were recommended by CE to FPL for specific application to St. Lucie Plant.

The following are the calculations which support FPL's use of RCS pressure plateaus that differ from the generic plateau of 1210 psia.

REALISTIC RCS PRESSURE PLATEAU CALCULATIONS

The RCS pressure following a small break LOCA exhibits a characteristic "pressure plateau" after reaching saturation condition as a result of a quasi-steady state energy balance which is coupled to a stable secondary heat sink. Under saturated conditions, RCS pressure corresponds to a pressure plateau as derived from the following equation (CEN-268, Supplement 1-P):

$$Q_{core} + Q_{RCP} - \frac{UA\ (T_{pri} - T_{sec})}{3600} = f[W_{leak}\ (P)] + g[W_{HPSI}(P)]$$

The computer code runs of CEN-268 Supplement 1-P, show primary pressure about 50 psia above secondary pressure. The purpose of this analysis is to explicitly demonstrate this order of magnitude of the RCS pressure plateau relative to Secondary (SG) pressure when realistic data are used in the calculations. By contrast, conservative data has been used in the calculation of CEN-268 Supplement 1-P, to determine the analytical set point of 1210 psia.

The terms on the right side of the equation both represent depressurization terms, since the leak term removes inventory from the RCS and since the HPSI term represents the pumping of cool HPSI water into the hot RCS. Consequently, a simplified calculation of the pressure plateau can omit these two terms, which are not large relative to the other terms, and the result of the calculation will then give a value slightly higher than the realistic pressure plateau.

The modified equation is therefore:

$$T_{pri} - T_{sec} = \frac{3600 \times (Q_{core} + Q_{RCP})}{UA}$$

Using the criteria from CEN-268, Supplement 1-P, of 4.3% decay heat for 2754 mwth, and 20 megawatts power for 4 RCP's, the 2 Q terms then sum to ~131,194 BTU's/sec. Also using a U factor of 600/BTU $ft^2/hr/^oF$ and the area for each generator of 90,232 ft^2, and crediting 2 steam generators whose secondary sides are covered (post reactor trip) gives a ΔT between primary and secondary of about 5^oF as shown below:

$$\Delta T = \frac{3600 \times (Q_{CORE} + Q_{RCP})}{UA}$$

$$= \frac{3600 \times 131{,}194}{(600)\ (2)(90232)}$$

$$= 4.36^{o}F$$

Hence, using the stated data, $\Delta T = 4.36^{o}F$. Converting ΔT into primary pressure depends on the secondary pressure used, P_{sec}; after the initial transient has taken place, this will be established at 900 psia for the Steam Bypass Control System (SBCS) or 1000 psia if MSSV's are used.

Noting from CEN-268, Supplement 1-P, that the RCS pressure plateaus are established at about the same time as P_{sec} stabilizes, it follows that a realistic calculation of pressure plateaus will use either 900 psia or 1000 psia depending upon what is controlling P_{sec}. The graph compiled from CEN-268, Supplement 1-P, data shows the evolution of the pressure plateau in terms of the secondary pressure at 900 psia. T_{sec} data corresponding to the above secondary pressures are:

900 psia, $531.95^{o}F$
and
1000 psia, $554.58^{o}F$

For these two cases, the plateaus correspond to primary temperatures of:

$T_{pri} = 531.95 + 4.36 = 536.31^{o}F$
and
$T_{pri} = 544.58 + 4.36 = 548.94^{o}F$,

Using the steam tables, the pressure plateaus are P_{sat} at each of two temperatures:

$536.31^{o}F$ gives 933.62 psia
$548.94^{o}F$ gives 1036.50 psia

In summary, the realistic pressure plateaus are about 35 to 40 psia above the corresponding secondary pressure. These values have been rounded up to 50 psia for use in this analysis.

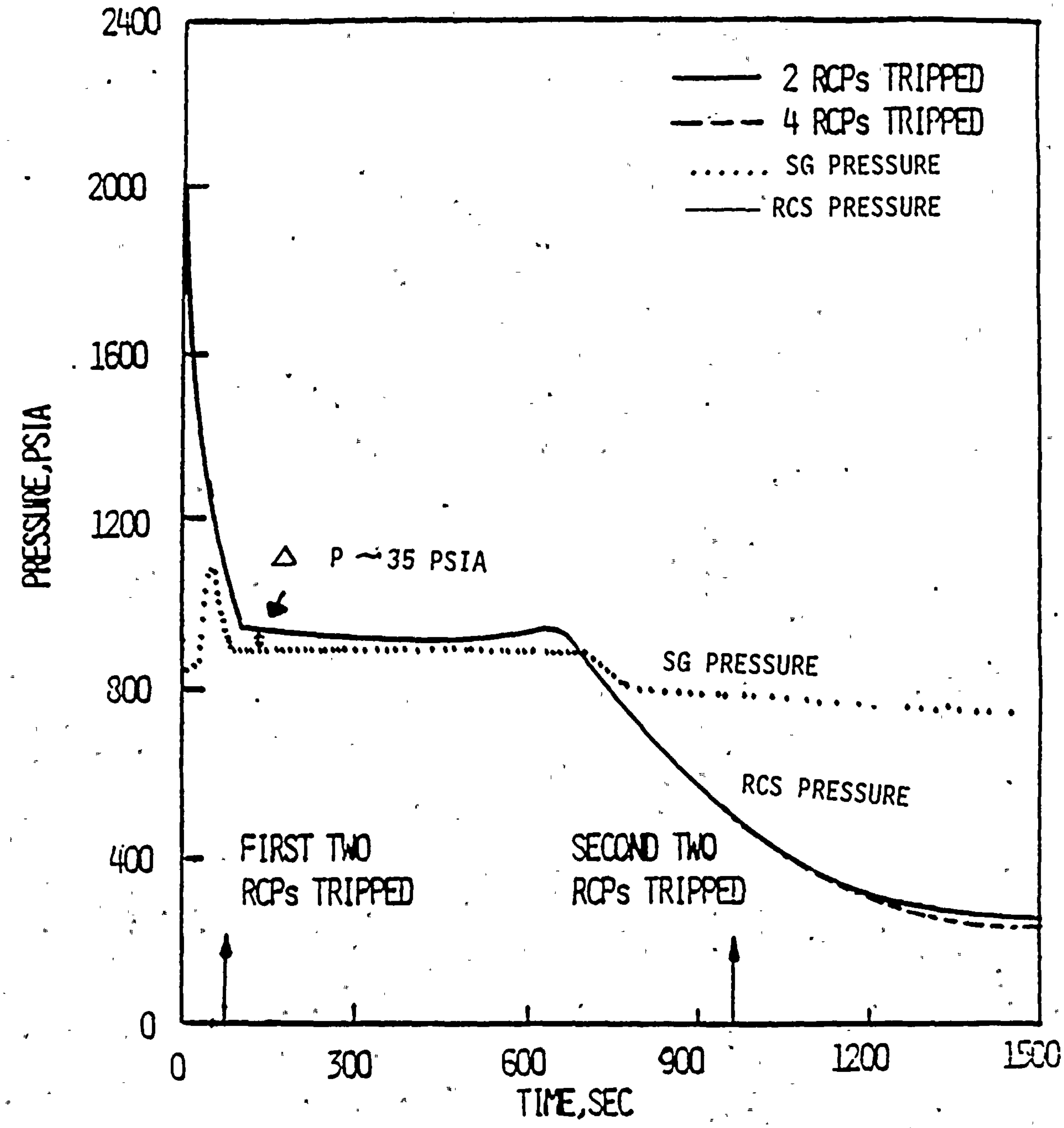
<u>Question Number 3</u>

The list of emergency operating procedures (EOPs) requiring the use of RCP trip guidelines provided by Florida Power & Light in response to GL 86-06, Item 4, did not include the EOP for loss of coolant accidents. Clarify why the EOP for LOCAs do not require the use of pump trip guidelines. Also, the list of procedures that provide direction for use of individual steam generators with and without RCPs did not include the EOPs for steam generator tube rupture or steam line break. Because a steam generator is lost in each of these accidents, clarify why these procedures do not provide guidance in this area.

RESPONSE

Site specific EOP's were written for PSL Units 1 and 2 with CE input. The Trip 2 Leave 2 (T2L2) strategy has been imbedded into the cumulative structure of the EOP's. Specifically, the strategy has been incorporated into the Standard Post-Trip Actions (SPTA), Loss of Coolant (LOCA), Steam Generator Tube Rupture (SGTR), Excess Steam Demand (ESD) and the Functional Recovery Guidelines (FRG). The complete strategy cannot be found in any single procedure, but would require the use of two different EOP's to be implemented. For example, guidance for tripping the first two reactor coolant pumps may be found in the SPTA, SGTR, and ESD procedures. Once a loss of coolant accident is diagnosed, the LOCA or FRG procedures will direct the operator to trip all or the remaining set of pumps. Additionally, if a LOCA is diagnosed from the outset and RCS pressure drops below the Safety Injection Actuation Setpoint, all four reactor coolant pumps are tripped. Tripping all pumps minimizes RCS inventory loss, and is therefore conservative. Thus, the EOP for LOCA need not require the use of a complete set of pump trip guidelines set forth by T2/L2. Furthermore, the T2L2 strategy is not applicable to the following EOP's: Reactor Recovery, Loss of Forced Circulation and Loss of Feedwater guidelines.

The following EOP's provide direction for the use of individual steam generators with and without the use of reactor coolant pumps: ESD, SGTR and FRG. ESD and SGTR procedures contain the T2L2 strategy for tripping the first two pumps as well as guidelines for tripping all four pumps when pump operating criteria is not met. The ESD and SGTR EOP's do <u>not</u> direct the operator to trip the second set of pumps based on T2/L2 strategy. Tripping the second pumps would indicate that a LOCA is diagnosed; therefore, the ESD and SGTR EOP's would direct the operator to the LOCA procedure where all reactor coolant pumps are tripped. The FRG assumes a worst case condition, LOCA, and contains guidelines for tripping all four reactor coolant pumps when primary pressure goes below 1300 psia. The LOCA procedure also contains guidance for tripping reactor coolant pumps when their operating criteria is not met.



0.1 FT^2 HOT LEG BREAK
STEAM BYPASS SYSTEM OPERATIONAL
RCS (PRESSURIZER) PRESSURE
AND
SG PRESSURE