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SUBJECT: Forwards addl info re loss of offsite power,in response to NRC 831012 ltr & 830926 telcon.

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FLORIDA POWER & LIGHT COMPANY

November 21, 1983
L-83-567

Office of Nuclear Reactor Regulation
Attention: Mr. James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miller:

Re: St. Lucie Unit I
Docket No. 50-335
Loss of Offsite Power -
<u>Additional Information Request</u>

In response to your letter of October 12, 1983, attached please find Florida Power & Light Company's answers to questions asked during a telecom with NRC on September 26, 1983.

Very truly yours,

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

JWW/RJS/cab

Enclosures

cc: J. P. O'Reilly, Region II
Harold F. Reis, Esquire

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ATTACHMENT 1

HISTORICAL BACKGROUND

Historical Perspective Applicable to All Questions

Two sets of analyses (each of which contained an analysis for both the Seized Rotor (SR) and the Loss of Offsite Power (LOOP) events) are germane to the present review of the LOOP at the NRC. The first is the SL1 Uprating submittal, which was submitted as Reference (7) and requested an increase in licensed operating power from 2560 to 2700 MWTThe second set, Reference (8), contains the LOOP analysis currently under review by the NRC. The second set was submitted in satisfaction of a commitment by Florida Power and Light (FPL) to provide a reassessment of the SR and the LOOP events to include additional assumptions which were not a part of the original SL1 design basis from the Final Safety Analysis Report (FSAR) but which were included in Post-FSAR Standard Review Plans (SRPs). This section provides some historical background to aid in understanding the differences in assumptions used. Many of the points which the NRC has asked FPL to clarify appear to arise from the assumption differences between the two analyses.

Cycle 4 operation at a licensed power of 2560 MWT began on May 10, 1980. The licensing basis for the Reload Safety Evaluation (RSE) supporting Cycle 4 operation at this power level included analyses for the SR and LOOP events which were consistent with the design basis approved in the FSAR in that no Worst Single Failure (WSF) assumption was made and Auxiliary Feedwater (AFW) was assumed to be manually initiated.

Reference (7) is the 2700 MWT Uprating submittal forwarded to NRC in November 1980. The various analyses contained in it used Cycle 4 parameters. It was both FPL's and the NRC's expectation that the review of Reference (7) would be completed sometime during the scheduled Cycle 4 and that the uprating would be accomplished "mid-cycle". Consistent with all prior 2560 MWT analyses, the new 2700 MWT SR and LOOP analyses in the stretch submittal did not account for a WSF, and both analyses assumed that auxiliary feedwater was manually initiated.

The accident at Three Mile Island (TMI) resulted in delays for all non-essential licensing activities by NRC. Review of FPL's Uprating submittal was consequently not completed until after Cycle 4 shutdown which occured on September 9, 1981. Because it was recognized that the first cycle of 2700 MWT operation would be Cycle 5, three analyses from Reference (7) (Main Steam Line Break, Excess Load, and Steam Generator Tube

Rupture) were reanalyzed incorporating Cycle 5 parameters and Post-TMI hardware modifications, notably the interim automatically initiated auxiliary feedwater system which had, by this time, been installed at the plant. These analyses were submitted to NRC by References (9) and (10). (The interim system provides auxiliary feedwater to both steam generators after a three minute time delay.) An assessment of the applicability of the other Cycle 4 analyses to Cycle 5 operations was submitted as Reference (11).

Prior to NRC approval for operation at 2700 MWT, FPL commited to submit two additional reanalyses (the SR and the LOOP) with additional assumptions not contained in the Unit 1 design basis but consistent with SRPs in use at that time. Cycle 5 operation at 2700 MWT commenced on December 2, 1981.

Reference (8) forwarded these reanalyses for SR and LOOP. The LOOP analysis contained in Reference (8) is the subject of the current review. These analyses differed from all prior SL1 submittals in that NRC had requested assumptions in accordance with recently updated Standard Review Plans (SRPs), i.e., the Seized Rotor also assummed a LOOP and a WSF; and the LOOP event assumes a WSF. In both cases the WSF was identified to be the failure of the ADV in the open position. Since Cycle 5 was already in progress, Cycle 5 parameters were used. Additionally, in order to envelope the effects of an anticipated plant hardware change, the upgraded "SMART" automatic AFW system was assummed for Reference (8), rather than the interim automatic AFW system currently installed.

ATTACHMENT 2

SPECIFIC QUESTIONS
CONCERNING THE LOOP ANALYSIS FROM TELECON WITH NRC
(Paraphrased in some cases for clarity)

NRC Question

What is the Worst Single Failure (WSF) for the LOOP event?

Answer

The WSF for the LOOP event has been determined to be the failure of an Atmospheric Dump VALVE (ADV) in the open position.

Such a failure could be postulated based on mechanical failure, electrical failure, failure of an automatic ADV control system (should one be used) or operator error. Failure of the ADV in the open position maximizes steam release from the secondary side and therefore also maximizes radiological dose.

The failure of the diesel generator to start was also considered as a candidate for WSF but was rejected since no equipment supplied with power by the diesel (which would then be unavailable if it failed) increases the radiological dose as significantly as failure of the ADV. Overpressure is not a concern from a loss of the diesel since overpressure criteria have been shown to be met by both SL1 and SL2 for complete station blackout (including diesel generator failures) as part of licensing actions for SL2.

NRC Questions

Are ADV's in "AUTO" mode or not? Why? Tech Specs?
What is "AUTO" mode for ADV's?
What is the setpoint?
Why does the unaffected ADV open at 2.08 seconds?

Answer

This analysis assumes automatically controlled ADVs, even though no such system is in use at SL1. ADVs at SL1 are only operated manually. (An automatic ADV system was included in the SL1 design but has never been used). The mode of operation for ADVs is not mentioned in Technical Specifications.

The reason for assuming a mode which is not used was to provide an analysis which conservatively enveloped both existing plant operation and anticipated future changes. At the time of the LOOP submittal, hardware and procedural changes to make an automatic ADV system at SL1 were under serious consideration for the upcoming Cycle 6.

For this analysis, one of SL1's two ADVs was assumed to fail in the open position at t=0. This failure is a postulated initial condition chosen to maximize the potential for fuel damage and radiological release and is not a predicted consequence of the progression of the events during the transient. This assumption conservatively envelopes the effects of either the failure of an automatically operated ADV system, an operator error, or a mechanical failure since the excessive cooldown starts immediately. (An automatic system, actuated by an increasing secondary pressure would not function (or fail) until later in the transient.)

The second ADV was allowed to function in accordance with the assumed automatic program, wherein the ADV will open at 970 psi increasing and reseat at 970 psia decreasing. The Reference (8) analysis predicts the opening of the second ADV at 2.08 seconds. Assuming automatic control of the second ADV is also conservative. It provides for greater and earlier cooldown than the Main Steam Safety valves (which have a higher opening setpoint).

NRC Questions

The Reference (8) analysis assumes that the ADV block valve is shut at 30 minutes.
Reference (7) assumed 15 minutes.
Why is there a difference?
What happens if the block valve is not closed at 30 minutes?

Answer

The assumed operator response time to manually block the failed ADV was conservatively increased from 15 to 30 minutes to demonstrate safe plant operation despite additional delays and also to be consistent with other submittals of the same time period.

FPL considers that 30 minutes is sufficient time for operators to recognize and block a failed ADV. Consideration of effects of failing to block the ADV within this period are thus unnecessary.

NRC Question

What are the assumptions concerning the Auxiliary Feedwater System ? Reference (7) says manual while Reference (8) says automatic.

Answer

For AFW, it was decided that the finalized design for the upgraded "SMART" AFW system would be assumed, even though its installation was not scheduled until Cycle 6. This was based on analysis results for the sister unit, SL2, which had a functionally similar "SMART" AFW system. The SL2 results showed that the radiological doses from events in which an ADV is assumed to fail are maximized with upgraded "SMART" AFW systems.

The designs of the interim AFW system currently installed and the proposed "SMART" system differ significantly. An Auxiliary Feed Actuation Signal (AFAS) will be generated on low steam generator level. The interim system delivers full AFW to both steam generators with a 3 minute time delay. The upgraded "SMART" system has additional logic which senses high differential pressures between either the two steam headers or the two feedline headers, indicative of a steamline or feedline break, respectively. If the differential exceeds a setpoint, the steam generator on the lower pressure is considered "faulted" and receives no AFW.

The important effect of the new system to both of the Reference (8) analyses (SR and LOOP) is that the failed ADV behaves similar to a steamline break, in that the differential between the two steam header pressures exceeds the setpoint, and AFW to the steam generator (SG) on the affected side is shut off. Subsequent dryout of the steam generator occurs and the dilution of assumed primary to secondary leakage (as accounted for by the iodine partition factor) is lost. When the partition factor rises from 0.1 to 1.0, the event's predicted offsite doses are maximized.

The setpoints and delays for the upgraded "SMART" AFW system and the associated timing for AFW flows evident in the sequence of events table in Reference (8) were based on the vendor's setpoints projected at the time of the submittal. Hardware issues unrelated to the analysis coupled with FPL's desire to review performance of the now operational SL2 "SMART" AFW system have resulted in a postponement of the SL1 installation until Cycle 7 (1985), and a reevaluation of all setpoints and delay times for the system.

The Reference (8) submittal remains conservative, however, because simply having the upgraded "SMART" system, with the setpoints used, results in predictions of steam generator dryout on one side, and the resulting loss of benefit from the partition factor. The interim system now installed would be predicted to supply AFW to both sides, without shutoff, and without the loss of the partition factor benefit.

A functionally similar system to the one used for this analysis is described in Section 10.4.9 of the St. Lucie Unit 2 (SL2) FSAR, Reference (6).

NRC Question

The NRC SER for Amendment 48 to the SL1 Operating License agreed that an initial SG pressure of 909 psia was conservative for the LOOP because it maximized the initial steam release. The Reference (8) analysis uses 900 psia. Is this still conservative?

The steam release numbers for the Reference (7) and Reference (8) analyses (shown below) are different. Explain.

	Reference (2)	Reference (3)	%
Release thru MSSV's 0-2 Hrs [lbm]	163000	16341	-89.9%
Release thru ADV's 0-2 Hrs [lbm]	590000	742712	+25.9%
Total Release 0-2 Hrs [lbm]	753000	759053	+ 0.8%
Total Release in Cooldown to 325 F [lbm]	903000	913780	+ 1.2%

Answer

For Reference (8), an ADV is assummed to fail open at t=0. This assumption results in significantly greater steam release in the early seconds of the transient over that which was seen for Reference (7), in which Main Steam Safety Valves (MSSVs) cycled only after steam pressures increased above their opening setpoint and ADV's were opened under operator control much later in the transient. Despite the greater steam release earlier in the transient, DNB limits are not breached, no fuel damage occurs, and radiological release remains determined by maximum Technical Specification limits on primary coolant activity. The 9 psi difference in the steam generator pressure therefore has no significance.

Differences do exist in reported mass integrated flows, primarily because the assumption of a failed ADV redistributes the flow path of secondary fluid out of the system from the MSSV's to the ADV's and also causes greater mass outflow in the early seconds. However, the reported total outflow (from both MSSV's and ADV's combined) for the first two hours and for the entire event (down to 325 $^{\circ}$F) are only about 1% or less different, which is well within expected agreement for the Reference (7) versus Reference (8) cases. Both with and without ADV failure, the same energy must be removed from the system for cooldown over the long term. The slight differences in mass outflows can be considered to be due to the differences in the timing of the mass outflows and slight differences in enthalpies at those times.

Since no fuel damage is predicted for either case, Technical Specification activities are assumed and radiological doses are effectively equivalent.

NRC Question

Why did CEA worth change from 5.3% $\Delta k/k$ in Reference (7) to 5.6% $\Delta k/k$ in Reference (8)?

Answer

Cycle 4 parameters were used for Reference (7) vis-a-vis Cycle 5 parameters for Reference (8), since Cycle 4 was over and Cycle 5 was already in progress when the second submittal was made.

NRC Question

Why was a Doppler Coefficient multiplier of 0.85 used in Reference (8) vis-a-vis the 1.15 multiplier used for Reference (7)?

Answer

The value of the Doppler Coefficient was inadvertantly changed between the two analyses. However the vendor has indicated that for the LOOP event for SL1, the transient is not sensitive to this value. Results of a reanalysis would not be expected to change by more than at most a few percent. The ADV failure assumption has a far more significant effect than the Doppler Multiplier. The change is not considered significant.

Justification for this assessment may be seen by considering that the Doppler coefficient is only important if its effects (1) adversely affect the functioning of the RPS, or (2) significantly degrade margins to the SAFDL's.

For this event, reactor trip occurs as the result of the low flow trip, sensed by comparing loop flows measured by Differential Pressure Cells to the minimum setpoint. The RPS function in this event is therefore clearly independent of Doppler.

The Doppler coefficient will affect the magnitude of the slight power excursion after trip, but it can be seen from both Reference (7) and Reference (8) that this excursion is small. No SAFDL is breached. The multiplier used is therefore of no significant consequence.

NRC Question

Why did the system flow change to 138.3×10^6 lbm/hr vis-a-vis 133.8×10^6 lbm/hr for the LOOP event ?

Answer

This is a typographical error. The correct value for the LOOP analysis is 133.8×10^6 lbm/hr. A new Table 1 for this event is provided on the following page and should replace the Table 1 for the LOOP from Reference (8).

The value of 138.3×10^6 lbm/hr is correct for the SR analysis since, consistent with the Reference (7) SR analysis, Statistical Combination of Uncertainty (SCU) methods were used.

TABLE 1

LOSS OF AC AND STUCK OPEN ADV
KEY PARAMETERS ASSUMED IN STEAM RELEASE CALCULATIONS

Parameter	Units	Cycle 5
Initial Core Power Level	MWt	2754
Initial Coolant Inlet Temperature	F	551
Initial Core Flow Rate	10^6 lbm/hr	133.8
Initial Reactor Coolant System Pressure	psia	2300
Initial Steam Generator Pressure	psia	900
Initial Steam Generator Level	ft	36.2 above tube sheet
Low Flow Analysis Trip Setpoint	% of initial flow	93.0
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho$ / F	+0.5
Doppler Coefficient Multiplier	-----	0.85
CEA Worth on Trip	$\%\Delta\rho$	-5.6
Reactor Regulating System		Manual Mode
Steam Bypass System		Inoperative
Auxiliary Feedwater System		Automatic

NRC Question

As shown below, differences in both the values and the timing of peak system pressures were noted between the Reference (7) and (8) analyses. Can this be explained ?

	LOOP [Ref (7)] w/o ADV faiure	LOOP [Ref(8)] w/ ADV failure
Peak RCS Pres.	2534 PSIA @ 7.4 sec	2457 PSIA @ 3.8 sec
Peak S/G Pres.	1034 PSIA @ 6.4 sec	1093.5 PSIA @ 13.38 sec

Answer

Because several assumptions were changed between the two submittals, (e.g. ADV failure, physics parameters, etc.) it is difficult to attribute the above effects to a specific cause.

However, note that the failure of an ADV at t=0 will result in a greater energy removal from the primary system earlier in the transient which would tend to cause both the lesser peak primary pressure and the occurence of the peak earlier in the transient.

The behavior of the steam generator pressure can be explained in part by noting that the peak pressure occurs on the unaffected side, i.e. the side with the normal (closed) ADV. The steam flow from the affected side (with the open ADV) from the Reference (8) submittal would tend to create larger pressure and temperature imbalances between the two steam generators. The times for peaks would also be affected.

Changes in physics inputs (Cycle 5 versus Cycle 4) would also result in changes to both peak values and timing.

In both submittals, the values were derived from the vendor's NRC-approved CESEC code using models which have been previously used and reviewed. Additionally, since overpressure limits are not breached, since DNB limits are not breached, and since the major parameter of interest then becomes the offsite dose, dictated primarily by the failed ADV assumption, the behavior of primary and secondary pressure peaks is not of primary interest for this transient.

NRC Question

Reference (8) reports the actuation of Safety Injection. Reference (7) does not. What is happening to the primary at this time ?

Answer

The Reference (8) submittal reports Safety Injection actuation at its correct trip point as a normal consequence of the cooldown process.

Reference (7) omitted mention of Safety Injection, but identical behavior would be expected.

In both cases, Safety Injection occurs late in the transient and has no effect on predicted accident consequences.

NRC Questions

What are the references for Reference (8) ?

Answer

The list of references for Reference (8) was inadvertently omitted from the submittal.

The list of references from letter, following as Attachment (3) may be used. The numbering convention is identical. Mention is made in the Reference (8) submittal of present References (1) through (5).

NRC has not been forwarded a copy of Reference (1), a draft Reload Safety Evaluation (RSE) for Cycle 5 operation supplied to FPL by the fuel vendor, Combustion Engineering (CE). In lieu of this document, FPL had submitted Reference (11). The important point is that the Reference (8) analysis was performed using parameters of the Cycle 5 reload, and not those of Cycle 4. Table 1 of Reference (8) correctly summarizes the parameters used.

References *

1. Letter R. R. Mills (CE) to C. G . O'Farrill (FPL), 10/1/81, F-CE-7567

2. CENPD-107, "CESEC - Digital Simulation of a C-E Nuclear Steam Supply System", April 1974

3. CENPD-135-P, "STRIKIN II - A Cylindrical Geometry Fuel Rod Heat Transfer Program", August 1974

4. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology", February 1980

5. CENPD-183, "C-E Methods for Loss of Flow Analysis", July 1975

6. St. Lucie Unit 2 - Final Safety Analysis Report, Section 10.4.9 ("SMART" AFW system description)

7. Letter R. E. Uhrig to D. G. Eisenhut, 11/14/80, L-80-381 (Original stretch power submittal)

8. Letter R. E. Uhrig to D. G. Eisenhut, 8/31/82, L-82-381 (Current submittal under review)

9. Letter R. E. Uhrig to D. G. Eisenhut, 7/23/81, L-81-306 (Post-TMI MSLB analysis)

10. Letter R. E. Uhrig to R. A. Clark, 9/4/81, L-81-388 (Post-TMI Excess Load and SGTR)

11. Letter R. E. Uhrig to R. A. Clark, 10/7/81, L-81-439 (Applicability of prior analyses to Cycle 5)

12. Letter J. R. Miller to R. E. Uhrig, 10/12/83

*References (1) through (5) are also the missing references for Reference (8).