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SUBJECT: Forwards response to request for addl info re use of Topical Rept XN-NF-86-132, "St Lucie Unit 1 LOCA-ECCS Analyses W/15% Steam Generator Tube Plugging." Approved ECCS evaluation model used.

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FPL

MAY 29 1987

L-87-222

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

Gentlemen:

Re: St. Lucie Unit 1
Docket No. 50-335
Linear Heat Rate

By letter L-86-522, dated December 23, 1986, Florida Power & Light Company (FPL) submitted a proposed license amendment to change the allowable peak linear heat generation rate of St. Lucie Unit 1 Technical Specification Figure 3.2-1, Allowable Peak Linear Heat Rate vs. Fraction of Core Height, to 15 kw/ft for all axial elevations for all times in core life.

By letter dated April 3, 1987 (E. G. Tourigny to C. O. Woody), the NRC Staff requested additional information concerning FPL's use of "St. Lucie Unit 1 LOCA-ECCS Analyses with 15% Steam Generator Tube Plugging", XN-NF-86-132, Exxon Nuclear Company, November 1986. The Staff stated that it required this information to continue its review of the proposed license amendment.

Attached is FPL's response to this request.

If further clarification is required on this response, please contact us.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy

COW/EJW/gp

Attachment

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

A001

REQUEST FOR ADDITIONAL INFORMATION ON ST. LUCIE UNIT 1 LOCA-ECCS ANALYSIS WITH 15% STEAM GENERATOR TUBE PLUGGING, NOVEMBER 1986

Ref.: (1) "St. Lucie Unit 1 Revised LOCA-ECCS Analysis with 15% Steam Generator Tube Plugging", XN-NF-85-117, Exxon Nuclear Company, Richland, WA 99352, November 1985.

(2) "St. Lucie Unit 1 Revised LOCA-ECCS Analysis with 15% Steam Generator Tube Plugging - Break Spectrum and Exposure Results", XN-NF-85-117, Supp. 1, Exxon Nuclear Company, Richland, WA 99352, December 1985.

(3) "St. Lucie Unit 1 LOCA-ECCS Analysis with 11% Steam Generator Tube Plugging", XN-NF-86-23, Rev. 1, Exxon Nuclear Company, Richland, WA 99352, March 1986.

(4) "St. Lucie Unit 1 LOCA-ECCS Analysis with 15% Steam Generator Tube Plugging", XN-NF-86-137, Exxon Nuclear Company, Richland, WA 99352, November 1986.

1. NRC Comment

It is not clear from the report that the approved ECCS evaluation model (SER to ANFC (ENC) dated 7/8/86) was used. Please affirm that the approved model was used.

Response

The approved ECCS evaluation model (SER to ANF (ENC) dated July 8, 1986) was used. Reference 5 in XN-NF-86-137 should read XN-NF-85-16(P)(A) rather than XN-NF-85-16(P), and Reference 6 in XN-NF-86-137 should read XN-NF-82-20(P)(A) rather than XN-NF-82-20(P).

2. NRC Comment

Approved model (7/8/86) requires three plant specific items:

(a) Leakage flow path justification
(b) Break size justification
(c) Axial power shape stored energy compliance (Section 1.A of Appendix K).

Please provide (a).

Paragraph 1.A of Appendix K considers (b) and (c) together. The logic presented in the report (Sec. 3.3) to justify break size selection is not convincing. If some previous model shows 0.8 to be the worst break, this does not mean that the new model will show this. The fact that you don't expect a change is an insuffcent basis. The model changes are extensive enough to warrant a limited (3 break) spectrum. Please address.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

Section 3.2.1 asserts that more power deposited in upper 1/3 of core produced higher PCT. No justification for this is provided for this model. Same is true of the assertion about "falling off slower", then concluding that a flat 15 kw/ft is justified. One does not follow from the other. An explanation is requested.

Response

(a) The approved methodology states that the leakage flow path from the upper plenum to the downcomer will be utilized only when it can be characterized and justified. This leakage path was only included in the St. Lucie Unit 1 reflood model. This leakage path between the core barrel and hot leg nozzles was characterized from plant drawings. There are two hot leg nozzles; however, leakage at only one hot leg nozzle was included in the reflood model for conservatism. This leakage flow path has a small effect on the reflood rate and PCT.

(b) The break spectrum was analyzed in References 1 and 2. Four double-ended-cold-leg-guillotine (DECLG) breaks were analyzed with break discharge coefficients of 0.4, 0.6, 0.8 and 1.0. Three split breaks were also analyzed with break areas equal to 0.4, 0.8 and 1.0 times the double-ended cross sectional pipe area. ANF analyses have consistently shown that the guillotine breaks are more limiting than split breaks. Also, ANF analyses using REFLEX and TOODEE2 have consistently shown that refill and reflood behavior differ little between the various guillotine breaks in a large break LOCA spectrum. Therefore, a LOCA break having fuel and cladding temperatures significantly higher than other break sizes at the end-of-bypass (EOBY) will aso have the highest peak cladding temperature (PCT) during reflood.

Preliminary break spectrum calculations out to EOBY were performed and documented in Reference 1. These calculations clearly showed the 0.8 DECLG break to be the limiting break size. Complete break spectrum calculations out to PCT were performed and documented in Table 2.2 of Reference 2 at peak LHR of 15 kw/ft at 0.6 of core height and an average steam generator tube plugging level of 15% to confirm that the 0.8 DECLG was the limiting break. The results are as follows:

	0.4 DECLG	0.6 DECLG	0.8 DECLG	1.0 DECLG	0.4 SPLIT	0.8 SPLIT	1.0 SPLIT
PCT (^{o}F)	1983	2064	2188	2108	1746	1911	1961
Peak Power Node Ave. Fuel Temperature at EOBY (^{o}F)	1460	1506	1666	1530	1395	1368	1369
Peak Power Node Cladding Temperature at EOBY (^{o}F)	1341	1357	1433	1377	1245	1217	1210

References 1 and 2 provided support for an LHR limit of 13.4 kw/ft at 0.81 of core height.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

Several changes were later made in the LOCA/ECCS analysis to support a higher LHR at a relative core height of 0.81. These changes are listed in Table 2.1 of Reference 3, and include changing the average steam generator tube plugging level from 15% to 11%, an increase in the initial secondary liquid mass, a decrease in accumulator line resistance, an increase in initial containment temperature, noninstantaneous as opposed to instantaneous isolation of the secondary, and an increase in core cross flow resistance. Calculations are performed at a peak LHR of 14 kw/ft at 0.81 of core height out of EOBY for the previously identified limiting break (0.8 DECLG) and the second most limiting break (1.0 DECLG) to confiirm that the limiting break was not changed. The original 0.4 DECLG and 0.6 DECLG break results were considered to be far enough removed from the limiting break, and the changes mentioned above were considered to have a small enough effect during blowdown that it would preclude the 0.4 DECLG and the 0.6 DECLG breaks from becoming the limiting break. The results at EOBY for the 0.8 DECLG and 1.0 DECLG breaks are shown below:

	0.8 DECLG	1.0 DECLG
PCT (^{o}F)	2183	Not calculated
Peak Power Node Ave. Fuel Temperature at EOBY (^{o}F)	1522	1430
Peak Power Node Cladding Temperature at EOBY (^{o}F)	1380	1326

The fuel rod stored energy at EOBY was significantly higher for the 0.8 DECLG break, thus confirming that the 0.8 DECLG break was still the limiting break.

The only change in the present analysis documented in Reference 4 that would affect the blowdown behavior was reverting back to an average steam generator tube plugging of 15% from 11%. From past experience, a 4% increase in tube plugging will have a small effect on PCT and a minimal effect on the system response during blowdown such that the limiting break will not change.

Use of the FCTF reflood correlations rather than the FLECHT correlations will not change the break spectrum for the following reason. For St. Lucie Unit 1, at a peak LHR of 15 kw/ft and using the FCTF correlations, an early rupture occurs at the peak power node prior to beginning of core recovery. The ruptured node is also the PCT node. Therefore, as stated earlier, the break size with the highest fuel rod stored energy at EOBY will be the limiting break. Since the FCTF correlations are not used until after the beginning of core recovery and do not affect blowdown behavior, use of the FCTF correlations will not change the limiting break.

(c) The purpose of Section 3.2.1 of Reference 4 was to describe the procedure for selecting a bounding axial power shape. The bounding axial power shape will be one that is peaked the highest in the core and has the flattest profile immediately downstream of the peak LHR in the region where PCT is expected to occur. The flatter axial profile produces a higher LHR downstream of the peak LHR, and therefore, a higher PCT if the PCT happens to occur downstream of the peak LHR location. The flatter shape downstream of the peak LHR will also typically have slightly more integrated power in the upper 1/3 of the core, although it is the higher LHR that may produce a higher PCT. The Cycle 7 EOC shape was determined to have the flattest profile downstream of the peak LHR location and was, therefore, selected for use in the analysis. In the Reference 4 analysis, it turned out that the peak LHR location was also the PCT location such that the LHR downstream of the peak did not directly affect the PCT. However, this was not known prior to running the calculations when the limiting EOC shape was being determined.

The axial power shape used in the analysis is expected to bound any potential realistic shape that may occur in the reactor. Therefore, a flat 15 kw/ft limit can be drawn independent of core height without affecting plant operation.

3. NRC Comment

In Table 2.1 core wide M-W reaction is 1% at 120 sec. Obviously it would be greater than 1% after 120 sec., thus violating that requirement of 50.46. Please explain and modify your submittal as necessary.

Response

A < sign was inadvertently left off in front of the 1% value. The hot pin metal-water reaction was 0.612% for the X/L=0.77 case and 0.485% for the X/L=0.85 case. Therefore, the core wide metal-water reaction is << 1%.

4. NRC Coment

Section 3.2.1 says, "Consistent with the methodology described in Section 3.0..." Nothing is described in 3.0. Please explain.

Response

The first sentence in Section 3.2.1 should read "Consistent with the methodology described in Section 3.2,...."