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EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards response to NRC 800620 ltr requesting info re effect of stagnant upper head fuel region w/structural heat included.For most limiting cases,head area void does not adversely affect any limiting safety criteria.

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FLORIDA POWER & LIGHT COMPANY

February 9, 1981
L-81-43

Director 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 Unit 1
Docket No. 50-335
Natural Circulation Cooldown

Please find attached our response to Question 3 of your letter dated June 20, 1980 regarding the effect of a stagnant upper head fuel region with structural heat included. We have reanalyzed the FSAR Chapter 15 design bases events which result in a depressurization of the primary system and have concluded that for the most limiting cases, the void in the head area does not adversely affect any limiting safety criteria.

Very truly yours,

Jack E. Vessely for

Robert E. Uhrig
Vice President
Advanced Systems and Technology

REU/JEM/ras

Attachment

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

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Question

Provide the results of a review of the events analyzed in Chapter 15 of your FSAR and either (1) show that the considerations of a stagnant upper head fluid region with structural heat included does not alter the analyses presented; in particular, events which depressurize or required depressurizing the primary system (e.g., stuck open turbine bypass valve, steam generator tube rupture), or (2) revise your analyses as necessary to properly account for the upper head fluid and structure temperatures and resubmit the results.

Response

The St. Lucie Unit I FSAR Design Basis Events (DBEs) which cause a depressurization of the primary system to or below SIAS are:

1) Excess Load Event
2) Steam Generator Tube Rupture Event
3) Steam Line Rupture Event

These DBE's were reanalyzed to determine the effect of a stagnant reactor vessel upper head region with metal structure heat transfer on process parameter variables. For the reanalysis, the version of the CESEC code used explicitly accounts for the voiding that takes place in the upper head region due to thermalhydraulic decoupling between this region and the upper plenum region subsequent to tripping of the reactor coolant pumps (RCPs). Voiding due to flashing as well as metal structure heat transfer (boiloff) was considered in the analysis.

One of the significant assumptions made in the reanalysis is the tripping of all RCPs about five seconds after the generation of the Safety Injection Actuation Signal (SIAS). The tripping of the RCP's causes thermalhydraulic decoupling of the upper head region and is characterized by progressively decreasing flow to this region from the upper plenum region. At the onset of natural circulation conditions, this flow is set equal to zero. Automatic actuation of auxiliary feedwater flow and delivery to the SG's three minutes after a low steam generator level reactor trip signal was assumed in the analysis. The key results and conclusions of the reanalysis are presented below for each DBE.

Excess Load Event

The reanalysis of the Excess Load event indicated the following.

The Excess Load event resulting in the maximum Reactor Vessel Upper Head (RVUH) voiding is the instantaneous opening of all steam dump and bypass valves at full power. The addition of auxiliary feedwater (AFW) increases and prolongs RCS cooldown thereby enhancing RVUH voiding. However, since AFW initiation occurs after the time of maximum voiding the peak void fraction is unaffected.

RCP trip results in a proportionately reduced RVUH coolant flow until natural circulation is established at which time all flow to the closure head is assumed to terminate. The decreased coolant flow inhibits RVUH cooldown which raises the upper head saturation pressure and therefore increases RCS pressure during periods of voiding. The increased RCS pressure diminishes safety injection flow which reduces the mitigating effect of safety injection on primary coolant shrinkage.

The maximum RVUH void resulting from an Excess Load event is approximately 34% of the upper head volume. Since the steam bubble does not expand beyond the RVUH, primary coolant circulation is unaffected. Furthermore, the main effect of voiding in the RVUH is to reduce the rate of primary depressurization. Since this occurs after the time of MDNBR and PLHGR, the transient approach to DNBR limit is not affected and thus the parameters of primary concern in the FSAR and subsequent reload analyses are unaffected. The reduction in safety injection flow resulting from an increase in RCS pressure does not impact criticality considerations since the core always remains subcritical. Following termination of coolant flow to RVUH, cooldown is accomplished through an exchange of coolant between the RVUH and the core outlet plenum. This exchange is driven by the expansion and contraction of the steam bubble. Additional upper head cooling is accomplished by conduction across the upper guide structure.

In conclusion, void formation in the RVUH during an Excess Load transient does not adversely affect primary coolant circulation or the transient approach to SAFDL's. Therefore the conclusions drawn in previous analyses remain valid.

MDNBR - Minimum DNBR

PLHGR - Peak Linear Heat Generation Rate

SAFDL's - Specified Acceptable Fuel Design Limits

STEAM LINE RUPTURE

The Steam Line Rupture event, which results in the maximum void formation in the reactor vessel upper head (RVUH), is initiated by a circumferential rupture of a steam line at the steam generator main steam line nozzle at full power. Initial void formation occurs at 10 seconds in the RVUH as a result of the rapid cooldown of the reactor coolant system (RCS), and is further enhanced by the manual trip of the reactor coolant pumps on SIAS due to low pressurizer pressure. Prior to initiation of auxiliary feedwater (AFW), the RVUH void fraction reaches a maximum of 65% at 62 seconds. At approximately 140 seconds when the affected steam generator blows dry, the RCS cooldown is terminated and the steam bubble in the RVUH collapses.

AFW is initiated on low steam generator water level and enters the steam generators approximately 180 seconds after reactor trip. A second RCS cooldown is initiated by AFW entering the affected steam generator. A second void formation occurs in the RVUH at 251 seconds and increases to 37%. As for the FSAR analyses it is assumed that the operator is able to identify and isolate the affected SG anytime after 1200 seconds. This terminates the RCS cooldown. The primary coolant begins to swell due to decay and residual heat in the primary components. This, in conjunction with safety injection, causes the steam bubble to collapse and the pressurizer to refill. By 1800 seconds the RCS is solid and the pressurizer liquid region is re-established. Plant cooldown can be accomplished by using the intact steam generator.

The void formation in the RVUH during a Steam Line Rupture event does not adversely impact the conclusions (i.e., critical heat flux is not exceeded) reached in the FSAR and subsequent reload applications.

Steam Generator Tube Rupture Event

The reanalysis of the SGTR event indicated the following.

The modeling of the stagnant upper head region with metal structure heat transfer results in the formation of voids in this region. The void fraction in the upper head region peaks at about 44 percent during the transient and gradually decreases under the combined action of the HPSI flow and the controlled cooldown at the steam generators. The upper head voids completely collapse at about 2345 seconds. The duration of the voids depends on the rate of cooldown of the primary side and the HPSI flow rate.

The voids are predicted to occur only in the upper head and the pressurizer regions of the RCS during the transient.

The amount of voids predicted is not large enough to expand the steam bubble beyond the upper head region and to the elevation of the hot legs. Therefore, natural circulation cooldown of the RCS is not impaired.

The prediction of the upper head voids in the reanalysis does not alter the conclusions of the previous Cycle 4 analysis. This Cycle 4 analysis supplements the FSAR as the reference analysis for St. Lucie 1. The results of the reanalysis not only show insignificant impact on the off-site doses, but also demonstrate that the plant can be maintained in a stable condition by the collapse of the upper head voids in a timely manner through manual control of the cooldown rate.

Subsequent to collapse of the upper head voids, the plant is maintained in a stable condition, and the operator can bring the plant to the shutdown cooling entry conditions, by cooling down the RCS at a prescribed cool-down rate using the intact steam generator and the condenser.