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RECIP.NAME RECIPIENT AFFILIATION
NOVAK,T.M. Assistant Director for Operating Reactors

mAH

SUBJECT: Forwards stress analysis of drain & fill method for cooling reactor & responses to NRC 800708 ltr re potential effects of rapid RCS depressurization.No excessive stress levels occurred during normal cooldown.

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FLORIDA POWER & LIGHT COMPANY

October 17, 1980
L-80-343

Office of Nuclear Reactor Regulations
Attention: Mr. Thomas M. Novak
Assistant Director of Operating Reactors
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Novak:

RE: St. Lucie Unit 1
Docket No. 50-335
Natural Circulation Cooldown

Our letter L-80-277 of August 25, 1980 promised the results of additional engineering work to demonstrate that the drain-and-fill method can safely be used to cool the reactor during natural circulation cooldown. Attachment A presents the results of our NSSS vendor's stress analysis, which shows that the rapid refill and drain of the reactor vessel head does not cause stress levels in excess of those occurring during a normal cooldown of 100°/hr. This supplements our response to Question 1 from your July 8, 1980 letter.

Our letter L-80-306 of September 16, 1980 promised a response to questions 2a and 2b from your July 8 letter concerning potential effects of the rapid RCS depressurization. Attachment B contains the response to those questions.

Very truly yours,

Jack E. Vessely for

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/PLP/ras

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

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ATTACHMENT A

RE: St. Lucie Unit 1
Docket No. 50-335
Natural Circulation Cooldown

A combined thermal/stress analysis was performed to determine the range of stresses produced in the reactor vesssel head due to voiding of the upper head region. In performing the thermal analysis, "worst case" assumptions were made in defining the fluid temperature and initial vessel temperatures. The assumptions which define the thermal transient are as follows and illustrated on Figure 1.

1. The upper head is drained down to the upper guide support structure plate in 20 minutes with both fluid and metal temperatures remaining at 600°F.

2. The water level is held at this height for 40 minutes with a fluid temperature of 300°F.

3. The head is refilled over a 20 minute period with 300°F water.

4. The upper head remains filled with 300° water for a period of time.

5. The heat transfer coefficient for the water is large (H=500 Btu/ft^2 - hr-°F).

6. The heat transfer coefficient for steam is very small (H=0. Btu/ft^2-hr-°F).

Temperatures calculated for this transient were applied to a stress analysis model. The results of this analysis indicate that the highest stresses occur in the "knuckle" region of the head near the inside radius. The magnitude of stresses produced for this transient were found to be no more severe than the stresses occurring during a normal cooldown of 100°F/hr.

In addition the results of this analysis demonstrate that:

1. A more rapid refill of the head does not cause higher stresses since the thermal conductivity through the reactor vessel wall is the limiting heat transfer mechanism.

2. The water level holddown time does have an effect on the stresses in the head. Longer holddown times decrease the stress in the "knuckle" region because of axial heat flow which removes heat from the head.

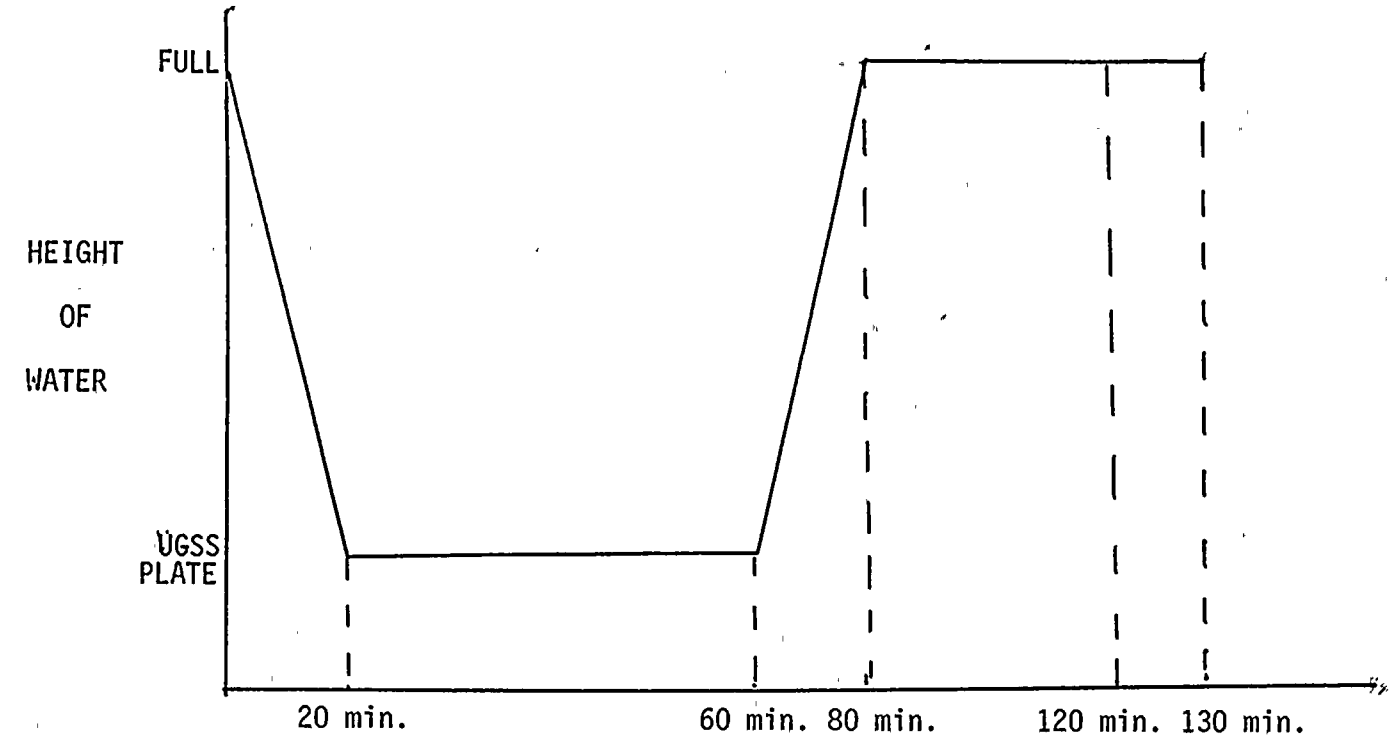
3. The thermally-induced stresses in the nozzle region of the reactor vessel are small in comparison to the stresses due to pressure loading only.

4. The deformations and rotations in the control element drive mechanism nozzles are negligible due to the thermal transient.

5. No separation occurs at the O-ring seal region of the flange, hence, no leakage occurs.

FORCING FUNCTION USED FOR

THERMAL TRANSIENT



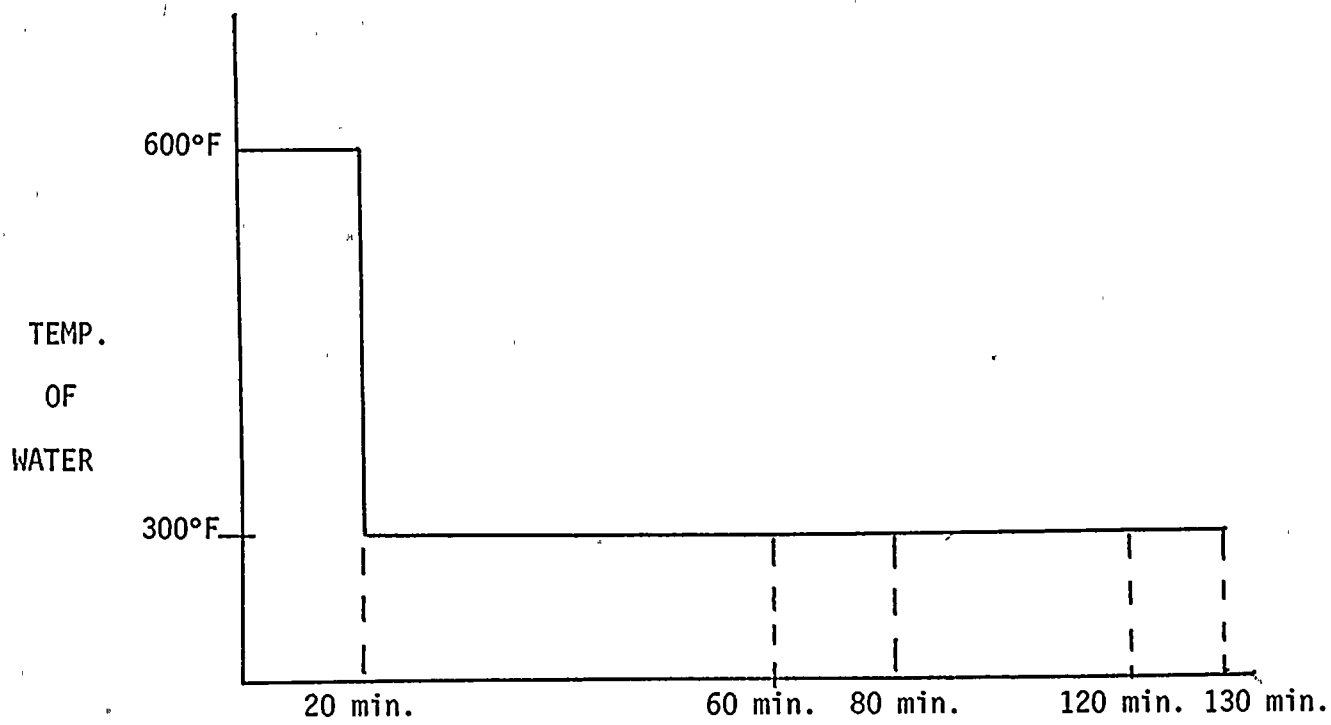


Figure 1

ATTACHMENT B

RE: St. Lucie Unit 1
Docket No. 50-335
Natural Circultaion Cooldown

QUESTION 2a:

Discuss the consequences, had a more rapid depressurization occurred. Could the steam bubble expand to the elevation of the hot legs? If so, what could be the consequences?

RESPONSE 2a:

Complete voiding of the entire region above the reactor vessel hot legs is not anticipated. However, a steam bubble could expand to the top of the hot legs if a sufficiently rapid depressurization were allowed to continue. The amount of depressurization required for this expansion to happen increases with time as the initially hot water in the reactor vessel upper plenum/upper head region cools with time. The consequences of steam entering the hot legs are minimal since:

1) The voids would be rapidly collapsed by thermal contact with the subcooled loop water.

2) If a steam void did proceed into the inlet side of the steam generator tubes, this would enhance natural circulation due to the resulting increase in driving head.

3) The condensation rate of a steam void in the steam generator tubes is more rapid than in the subcooled loop so that significant degradation of the natural circulation process due to steam bubbles in the tubes will not occur.

4) Natural circulation two phase heat transfer in the steam generator tubes has been demonstrated to be an adequate cooling mechanism even where large amounts of voiding are present; this is described in CEN-114, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply System, July 1979, which was transmitted to the NRC as part of the response to NUREG 0578.

QUESTION 2b:

The operators depressurized the plant as fast as they did due to a concern of pump seal failure. Evaluate the consequences of the seals failing on all four pumps.

RESPONSE 2b:

While there was a concern for RCP seal failure, the plant cooldown was not expedited, but rather,was conducted at the normal rate (50°/hr). The simultaneous failure of all four RCP seals is a

highly improbable event, and it is especially improbable due to loss of cooling water. Additionally, recent hot standby tests of an RCP seal cartridge of the type utilized by St. Lucie Unit 1 demonstrated that with the RCP's stopped, the seal cartridge will continue to function as a pressure containing component for an extended period of time without cooling water.

The test was conducted with the test loop temperature and pressure at primary loop normal hot operating conditions. The seal cooling water was shut-off and as anticipated the temperature in the vapor seal began to rise. This temperature rose to approximately 400° and remained there throughout the test. Leakage through the controlled bleed-off and the vapor seal was normal and the pressure drops across the seals remained fairly constant. The test goal was four hours without serious leakage rates, but to be continued for 24 hours if conditions permitted. The seal test actually ran for about 56 hours and the seals had not "failed" at the end of that time.

The results of the test and previous operating experience proved that the seals can withstand the loss of cooling water for an extended period of time, without failure.