P. O. BOX 013100, MIAMI, FLORIDA 33101
FLORIDA POWER & LIGHT COMPANY

September 9, 1975
L-75-434 Records Facilities Branch

Mr. Norman C. Moseley, Director
Office of Inspection and Enforcement,
Region II
U. S. Nuclear Regulatory Commission
230 Peachtree Street, N.W., Suite 818
Atlanta, Georgia 30303

50-335

Dear Mr. Moseley:

Re: Construction Incident Report, St. Lucie 1
Split Flange on Intake Cooling Water Line

On August 11, 1975, Florida Power & Light Company notified your office of a reportable incident under 10 CFR 50.55(e), in which a bolted flange on Intake Cooling Water Line I-3-CW-70 had split. The report on this incident has not yet been completed and is due to be submitted on September 10, 1975. Accordingly, FPL requests postponement of this report until October 6, 1975. This extension was discussed with and noted by Mr. Foster of your office on September 9, 1975.

Yours very truly,

for Robert E. Uhrig
Vice President

REU:nch

cc: Jack R. Newman, Esquire

FLORIDA POWER & LIGHT COMPANY

September 4, 1975
L-75-426

Mr. Norman C. Moseley, Director
Office of Inspection and Enforcement, Region II
U. S. Nuclear Regulatory Commission
230 Peachtree Street, N. W., Suite 818
Atlanta, Georgia 30303

Dear Mr. Moseley:

Re: IE:II:MSK
Reactor Coolant System Overpressurization,
St. Lucie Unit No. 1

On August 12, 1975, letdown isolation valve #V2516 failed closed and permitted an 80 psi overpressurization of the Reactor Coolant System of St. Lucie Unit No. 1. The valve closure was caused by a broken electrical lug on HGA relay #94-3-159. A report of this incident, stating the circumstances, the analytical results, and the corrective action taken, is attached to this letter. In view of the fact that no damage occurred and that no extensive evaluation was required, FPL believes that this is not a reportable incident in accordance with 10 CFR 50.55 (e) (iii).

Yours very truly,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU:nch
Attachment

cc: Jack R. Newman, Esq.

ST. LUCIE UNIT NO. 1

REACTOR COOLANT SYSTEM OVERPRESSURIZATION

SEPTEMBER, 1975

TABLE OF CONTENTS

I Summary

1. Synopsis of Incident

On August 12, 1975; St. Lucie Unit No. 1 was undergoing hot functional testing, when letdown isolation valve #V2516 failed closed. The resultant associated transient caused the RCS pressure limitations, as shown in Figure 3.4-2a of the standard technical specifications to be exceeded by 80 psia. The failure of valve #V2516 was caused by a broken electrical lug on HGA relay #94-3-159.

2. Resolution of Damaged Equipment

The affected relay leads were relugged and tightened and the letdown isolation valve was returned to an operable status. No damage to the RCS was experienced.

3. Resolution of Occurrence

An analysis of the overpressurization condition determined that the overpressure transient was acceptable. See Section III for details. It is believed that failure of the electrical connetion on the relay lead was an isolated occurrence. The other factory-wired relay panels have been checked to assure lug tightness.

II Description of Incident

The failure of letdown isolation valve #V2516 on St. Lucie Unit No. 1 occurred on August 12, 1975, at 10:45 A.M. as the unit was undergoing hot functional testing. At the time of occurrence the Reactor Coolant System was under the following conditions:

1. Reactor coolant pressure at 210 psia as read on PI-1103 (scale 0-1600 psia)

2. Reactor coolant temperature at 105°F as read on pressurizer water phase temperature indicator TI-1101 (scale 1-700°F)

3. Charging and letdown system in operation with one (1) charging PP running.

4. Control element drive mechanism venting in progress.

The transient conditions following closure of V2516 were:

1. RCS pressure 600 psia as read on PI-1103.
2. RCS pressure approximately 660 psia as read on Heise gauge installed at vessel head
3. RCS temperature at 105°F as read on TI-1101.

Containment ambient temperature was 92°F as measured on recorder TR-25-1 (inlet temperature to the containment coolers).

The failure of V2516 occurred due to a broken lug connection on sealing relay #94-3-159, which is the solenoid relay associated with V2516. The plant's Instrumentation and Control group were in the process of checking lug tightness on the factory-wired relay panels on the RTGB, and in the process of removing the cover from relay #94-3-159, the letdown isolation valve (V2516) failed closed. Subsequent investigation revealed that the two lead lugs were broken off at the head of the lugs, and the relay cover was apparently holding the leads in place.

The transient associated with V2516 failing closed resulted in RCS pressure exceeding the pressure/temperature limitations as shown on Figure 3.4-2a in the standard technical specifications. Per Figure 3.4-2a the unit must be at greater than 156°F (RCS temperature) before exceeding 520 psia RCS pressure.

II. Corrective Action

1. Response by Field Personnel

Upon seeing the RCS pressure increase, the operator terminated the transient by securing charging to the RCS. The RCS pressure remained at approximately 600 psia for 2 to 3 minutes, at which time it was reduced to approximately 210 psia by venting via the vessel head vent.

The affected leads were relugged and tightened, and the letdown isolation valve was returned to an operable status at 1:40 PM on August 12, 1975.

2. Damage Investigation and Results

a. Relay Wiring

The relay and wiring involved was factory-wired at General Electric where the panels were fabricated. The lug used was an insulated ring tongue lug compressed onto a size #16 Vulkene flexible wire. It is probable that during the start-up check outs, that this lug was flexed each time the wire identification number was read and the lug finally broke at the junction of the ring and barrel when the relay cover was removed for inspection. The relay and the broken lug were inspected by FPL Electrical QA personnel on August 12, 1975. It was their opinion based upon the lack of any similar experience with the thousands of other similar factory-wired relays of this identical type, that

this occurrence was not a generic problem with either the lug or the relay, but an isolated instance.

b. Reactor Coolant System

On August 13, 1975, Combustion-Engineering, the NSSS vendor, initiated an evaluation of the overpressurization condition. The transient which occurred when V2516 failed imposed conditions on the RCS outside the limits specified in Figure 3.4-2a of the technical specifications. The particular limit violated was that of the lowest service temperature of 160°F as determined by NB-2332(b) of ASME Boiler & Pressure Vessel Code, Section III, 1971 Edition, Summer 1972 Addenda. The reference temperature, RT_{NDT}, upon which this 160°F lowest service temperature is based, is 50°F. A comparison between the reference (critical) intensity factor (K_{IR}) and the calculated stress intensity factors (K_{Im} and K_{It}) was made using the procedures of ASME Section III, Appendix G, in order to verify the acceptability of the overpressure transient.

1. Determine K_{IR}

$RT_{NDT} = 50^{\circ}F$

Temperature = $105^{\circ}F$. Assume $-10^{\circ}F$ instrument error and consider temperature relative to RT_{NDT}, $(T - RT_{NDT})$, to be (95°F - 50°F) = 45°F.

Enter Figure G-2110-1 at $+45^{\circ}F$ and determine K_{IR} = $\underline{\underline{50 \text{ KSI } \sqrt{\text{In.}}}}$

2. Determine K_{Im}

$\sigma_m = \frac{Pr}{t}$. Indicated pressure on Heise gauge was 660 psia. With 1/2% accuracy this pressure could have been 663 psia.

(a) 30" D Reactor Coolant Piping

$r = 15"$, $t = 2.5"$, $\sigma_y = 38$ KSI

$$\sigma_m = \frac{Pr}{t} = \frac{663 \times 15}{2.5} = 3980 \text{ psi} = 3.980 \text{ KSI}$$

$$\frac{\sigma_m}{\sigma_y} = \frac{3.980}{38} = .105, \sqrt{t} = 1.58$$

From Figure G-2114.1, find $M_m = 1.88$ and

$$M_m \times \sigma_m = 1.88 \times 3.980 = \underline{\underline{7.48 = K_{Im}}}$$

(b) 42" Reactor Coolant Piping

$r = 21"$, $t = 3.75"$, $\sigma_y = 38$ KSI

$$\sigma_m = \frac{Pr}{t} = \frac{.663 \times 21}{3.75} = 3.713 \text{ KSI}$$

$$\frac{\sigma_m}{\sigma_y} = \frac{3.713}{38} = .098, \quad \sqrt{t} = 1.94$$

From Figure G-2114.1 find $M_m = <1.88$ and
$M_m \times \sigma_m = <1.88 \times 3.713 = \underline{\underline{<6.98\ K_{Im}}}$

(c) Steam Generator Plenum

$r = 74.7"$, $t = 7.0"$, $\sigma_y = 70$ KSI

$$\sigma_m = \frac{Pr}{2t} = \frac{.663 \times 74.7}{2 \times 7.0} = 3.538 \text{ KSI}$$

$$\frac{\sigma_m}{\sigma_y} = \frac{3.538}{70} = 0.50 \quad \sqrt{t} = 2.64$$

From Figure G-2114.1 find $M_m = <<2.46$ and

$M_m \times \sigma_m = <<2.46 \times 3.538 = \underline{\underline{<8.70}}\ K_{Im}$

(d) Pressurizer

$r = 48.250$, $t = 4.875"$, $\sigma_y = 70$ KSI

$$\sigma_m = \frac{.663 \times 48.25}{4.875} = 6.562 \text{KSI}$$

$$\frac{\sigma_m}{\sigma_y} = \frac{6.562}{70} = .094, \quad \sqrt{t} = 2.21$$

From Figure G-2114.1 find $M_m = 2.08$ and

$M_m \times \sigma_m = 2.08 \times 6.562 = \underline{\underline{13.65\ K_{Im}}}$

(e) The reactor vessel temperature-pressure limits have been determined in accordance with Appendix G and are shown in Figure 5.4-1A of the St. Lucie Unit No. 1 FSAR. The pressure transient experienced did not violate the temperature-pressure restrictions as shown in this figure.

3. Since the transient was a pressure transient only, with the insulated system at a steady 105°F and the containment ambient at 92°F, the temperature difference through the

walls (ΔT_w) is insignificant and therefore the calculated thermal stress intensity, K_{It}, is insignificant.

4. Allowable pressure.

Using the criterion of Appendix G-2215, $2\ K_{Im} + K_{It} < K_{IR}$, and evaluating for the highest value of K_{Im}, as found for the pressurizer shell, we find that

$K_{IR} = 50$

$2\ K_{Im} = 2 \times 13.65 = 27.3$

$\underline{\underline{27.2 < 50}}$

Therefore, it can be concluded from this analysis that no stress intensities, such as to possibly propagate the conservatively assumed defects of Appendix G, Section G-2120, were encountered. It should be noted here that the value of $RT_{NDT} = 50°F$ is conservative and was used for this entire evaluation (expect for the vessel), not just the piping to which it is applicable.

3.. Action Taken to Prevent Recurrence

The other factory-wired relay panels have been checked to assure lug tightness.

V Safety Implications

There will be no effect on the safe operation of St. Lucie Unit No.1 as a result of this incident. Failure of the letdown isolation valve by the mechanism described is considered to be an isolated event. Operator action terminated the transient immediately after observation and brought the RCS within pressure limits shortly thereafter. An analysis of the overpressurization condition determined that the stress intensities experienced by the REC were acceptable in accordance with ASME Code, Section III, Appendix G.