UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-250-SP
	)	50-251-SP
FLORIDA POWER & LIGHT COMPANY	)	
	)	Proposed Amendments to
(Turkey Point Nuclear	)	Facility Operating License
Generating Units Nos. 3	)	to Permit Steam Generator
and 4)	)	Repairs)

AFFIDAVIT OF F. G. FLUGGER

My name is Frederick G. Flugger. My business address is 9250 West Flagler Street, Miami, Florida 33152. I am Manager of Plant Engineering Licensing, Power Plant Engineering Department, Florida Power & Light Company (FPL or Licensee). A statement of my professional background and qualifications has been previously provided with the "Affidavit of Frederick G. Flugger and H. H. Jabali on Contention 4A," which was attached to "Licensee's Answer Supporting NRC Staff Motion for Summary Disposition of Contention 4A" (April 17, 1981).

The purpose of this Affidavit is to respond to the Order of the Licensing Board dated July 28, 1981. This Order referred to the discovery, during preparations for the steam generator replacement, of a void in the concrete in the containment in the area adjoining and beneath the equipment hatch. It also referred to Section 4.1.2 of the Steam Generator Repair Report (SGRR), which states that the

steam generator lower assemblies (SGLAs) will be removed through the equipment hatch and that "No modifications impacting the integrity of the equipment hatch as a pressure boundary structure are contemplated at this time." The Board then posed the following questions:

1. How was the void in the containment wall discovered?
2. Did the discovery result from breaching of the containment wall, or in some other fashion?
3. If the wall was breached, why was the Licensing Board not informed?
4. Should the statements of Licensee in the SGRR be considered a commitment?
5. What has been the role of the NRC Staff in this matter?

Finally, the Order directed the Licensee and the NRC Staff to "provide the Board with full information concerning the apparent breaching of the containment wall at Unit 3, including answers to the questions raised in this Order."

Each of the Board's questions is discussed below.

1. <u>How was the void in the containment wall discovered?</u>

The void in the containment wall was discovered during removal and replacement of a portion of the equipment hatch

sleeve in accordance with Sections 3.2.5 of the SGRR. This section states:_/

> "The following structures or portions of structures will be removed to provide a path for the lower assembly (refer to Figures 3.2-4 through 3.2-6 for illustration):
>
> * * *
>
> i. A portion of the equipment hatch sleeve at elevation +30'-6". (A small section of the steel sleeve will be replaced with thicker steel to assure load transfer to the supporting concrete wall during ingress and egress of heavy equipment. The affected portion of steel to be replaced does not form a part of the containment pressure boundary nor affects the structural integrity of the containment.)"

_/ Copies of the relevant pages of the SGRR (pp. 3-9, 3-9a) are attached for convenience.

Figure 1 depicts the location of the equipment hatch sleeve and the location of the void. As can be seen from the figure, the equipment hatch sleeve varies in thickness. In order to assure load transfer during movement of heavy objects (such as an SGLA) through the equipment hatch during the repairs, FPL planned to replace the thin steel portion of the sleeve with a thick steel plate. During removal of a portion of the affected areas of the sleeve, FPL discovered the void in the concrete underlying the plate. The extent of the void was determined by probing, and a larger portion of the plate was subsequently removed to expose the entire void. The void is not extensive, it was not caused by the repair work, and it was probably formed during construction of Unit 3. It should be noted that upon completion of construction, the containment was subjected to and successfully passed, testing required by the NRC prior to initial operation. Thus, it is reasonable to conclude that the void does not impact upon the integrity of the pressure boundary.

As is apparent from Figure 1, removal and replacement of a portion of the thin section of the sleeve does not affect the containment pressure boundary. The containment pressure boundary consists of the internal containment steel liner, the gasketed equipment hatch closure door, and part of the thick section of the steel sleeve. The equipment hatch sleeve area involved in the removal and replacement of the

thin section of the sleeve is outside of the containment pressure boundary. Consequently, no portion of the containment pressure boundary was modified or affected by removal of the sleeve. The void is exterior to the containment pressure boundary, and neither the void nor procedures followed for repair of the void will affect the containment pressure boundary. Prior to return of Unit 3 to power, appropriate local leak testing will be conducted for areas affected by the steam generator repair activity to ensure integrity of the containment pressure boundary. See SGRR section 4.1.2.

The void is localized and not large; it will be repaired; the repair procedure is straight forward; and the void will in no way impact the SGLA repair activity. The NRC has been informed of the existence of the void and will be provided with the data necessary to disposition this matter.

2. <u>Did the discovery result from breaching of the containment wall, or in some other fashion?</u>

As described above, the discovery of the void did not result from a breach of the containment wall, but instead from removal of a portion of the thin section of the sleeve. There has been no breach of the structural integrity of the containment pressure boundary.

3. <u>If the wall was breached, why was the Licensing Board not informed?</u>

Since the wall was not breached, and the void was not caused by the repair work and does not impair the safety of the repairs, it was not thought necessary to inform the Board. However, as indicated in response to Question 5, below, the NRC Staff was advised.

4. <u>Should the statements of Licensee in the SGRR be considered a commitment?</u>

Yes. The SGRR provides the design and safety related bases for the repair that have been reviewed and accepted by NRC. It is the Safety Analysis Report for the repair project. Any deviation from the SGRR's design and safety-related bases would be reviewed by FPL in accordance with the Commission's regulations at 10 CFR § 50.59. Any change in the repair activity, as discussed in the SGRR, that involves an Unreviewed Safety Question must be approved by NRC.

FPL's statements in the SGRR that the Steam Generator lower assemblies would be removed through the equipment hatch and that no modifications would be made which would impact the integrity of the equipment hatch as a pressure boundary structure may be considered as design bases and are therefore commitments. Any deviations from these commitments (none are contemplated) would be dispositioned in accordance with 10 CFR § 50.59.

It should be noted that in addition to design and safety-related bases, the SGRR also contains details relating to the final design and implementing procedures. Such details are provided in the SGRR primarily to demonstrate the feasibility of the repair activity and to respond to NRC Staff queries. These details may be modified during development of the final design, e.g., to accomodate a design approach that results in a lower man-rem exposure, or that is more cost effective. Ordinarily such changes in detail do not involve an Unreviewed Safety Question and may be made in accordance with 10 CFR § 50.59. Thus, in all cases the detailed final implementation will comply with the commitments stated in the SGRR. <u>See</u> SGRR, p. 3-1.

5. <u>What has been the role of the NRC Staff in this matter?</u>

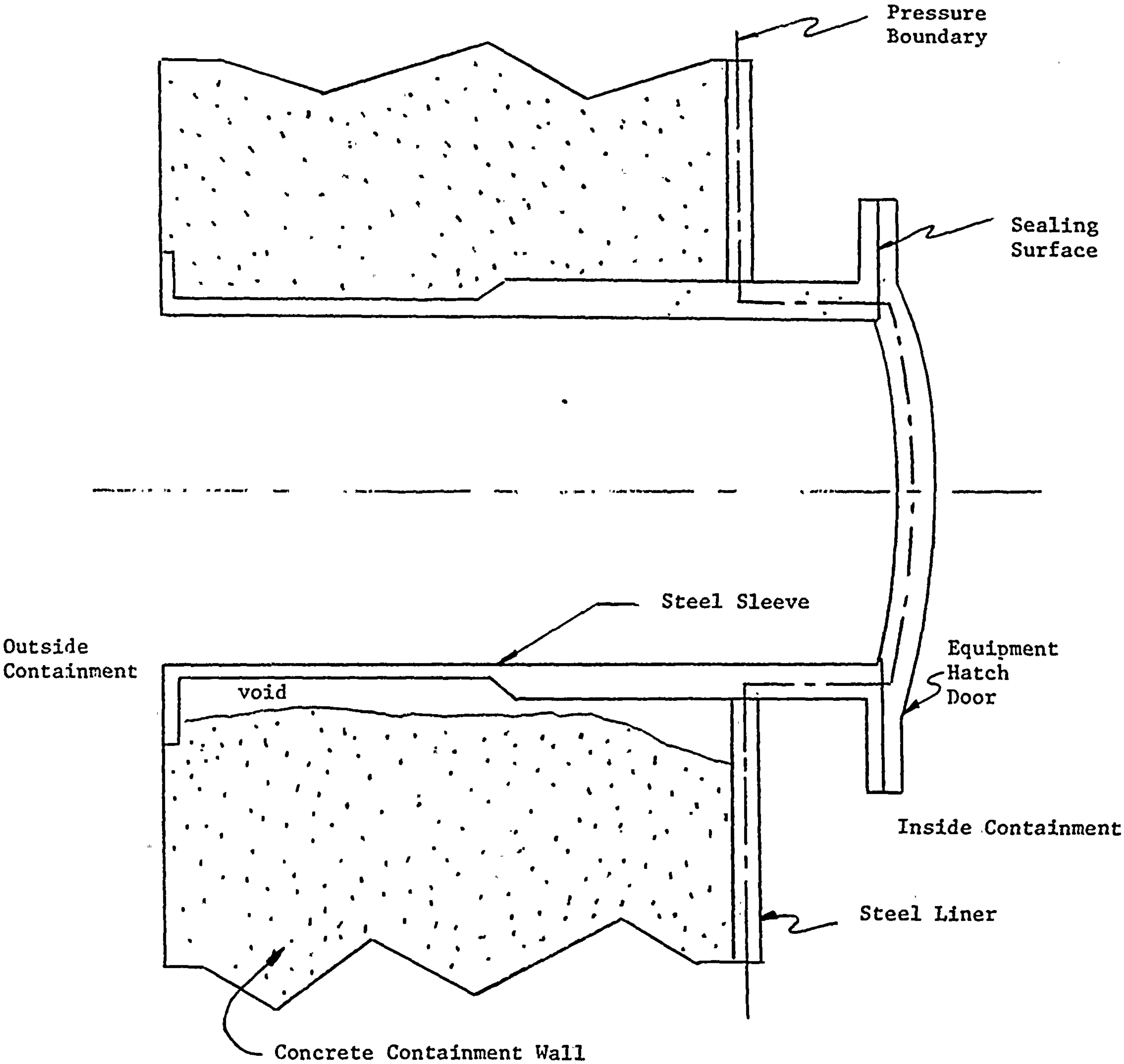
The void was first discovered on July 4, 1981, and a quality control and engineering evaluation process was initiated. By July 10, 1981, enough information was available to determine that the void was reportable. Therefore, by telephone on that day, FPL notified the Office of Inspection and Enforcement for Region II of the existence of the void. This was followed by the submission of a License Event Report on July 24, 1981. To the best of FPL's knowledge, the NRC Staff is reviewing this situation in accordance with its normal procedures.

* * *

In summary, a pre-existing concrete void was discovered during the course of repair work as described in the SGRR, this work did not entail a breach of the containment pressure boundary or cause the existence of the void, and the existence of the void will not impact the safety of the repair activity.

Figure 1

Diagram of Equipment Hatch Area*



*This diagram is not drawn to scale and omits components which are not relevant to the discovery of the void. It represents the configuration at the time of void discovery.

FURTHER AFFIANT SAYETH NOT

Date: August 5 1981

Frederick G. Flugger
Frederick G. Flugger

STATE OF FLORIDA)
) SS.
COUNTY OF DADE)

SWORN to and subscribed before me this 5th day of [illegible], 1981.

[illegible]
Notary Public
My Commission Expires: 12-05-82

g. Service air piping

h. Primary service water piping

Location of cut areas for reactor coolant system, main steam system, and
main feedwater system are shown on Figures 3.2-1, 3.2-2, and 3.2-3, 7
respectively. As appropriate the open ends of cut piping will be covered
to ensure cleanliness during the repair.

The governing overall code for the steam generator replacement shall be
the ASME Section XI, 1977 Edition with addenda through the summer of 7
1978.

3.2.5 Concrete and Structural Steel

The following structures or portions of structures within the containment will be removed to provide a path for the lower assembly (refer to Figures 3.2-4 through 3.2-6 for illustration):

a. A section of the steam generator "C" shield wall above elevation + 58' for Unit 4. A section of the steam generator "A" shield wall above elevating + 58' for Unit 3.

b. A section of the operating floor concrete at elevation + 58' including a steam generator "A" upper support embed for Unit 3 and steam generator "C" thrust beam for Unit 4

c. The removable secondary shield wall panels opposite the containment equipment hatch from elevation + 30'-6" to elevation + 58', and an additional width of secondary shield wall opposite the equipment hatch from elevation + 30'-6" to elevation + 58'

d. A portion of the floor framing and grating at elevation + 58' above the equipment hatch

e. A portion of the floor framing and removable floor slabs at elevation + 30'-6" at the low point of the equipment hatch

f. The upper portion of the steel stairway near the equipment hatch opening

g. A reinforced grouted pad in the equipment hatch at elevation + 30'-6"

h. A portion of the truss system tie rods to allow for clearance of the temporary polar crane trolley. (The truss system was originally utilized in the construction of the containment and does not perform any structural related function at present.)

i. A portion of the equipment hatch sleeve at elevation +30'-6". (A small section of the steel sleeve will be replaced with thicker steel to assure load transfer to the supporting concrete wall during ingress and egress of heavy equipment. The affected portion of steel to be replaced does not form a part of the containment pressure boundary nor affects the structural integrity of the containment.) 7

3.2.5.1 Removal of Concrete Structures

Removal of bulk volumes of containment internal structural concrete will utilize equipment and techniques commercially available. The intent is

[illegible]